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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Radiation Detection and Measurement, contains eight modules: Statistics of Radiation Counting: Gas-Filled Detectors I: Ionization Detectors: Gas-Filled Detectors II: Proportional Counters: Gas-Filled Detectors III: Geiger-Mueller Counters: Scintillators, Semiconductors and Spectroscopy: External Personnel Dosimeters for Beta-Photon Monitoring: Neutron Detectors, Dosimetry and Activation Analysis: and Radiological Calibration and Standardization. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

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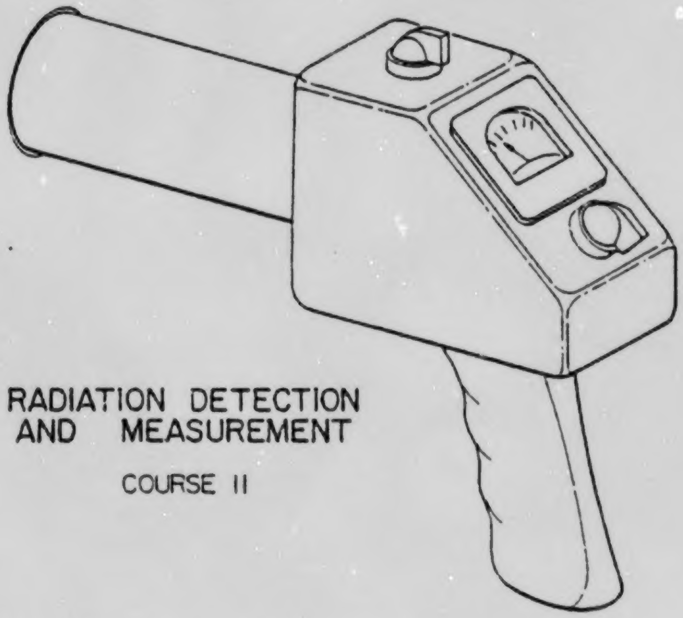
NUCLEAR TECHNOLOGY
Series

MODULE 11-1 STATISTICS OF RADIATION COUNTING. -



RADIATION DETECTION
AND MEASUREMENT

COURSE II

CE026652

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COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

In most technical applications of radioactivity in science and industry, measurements made are proportional to the radiation emitted from a radioactive source. This statement means that the measurements, which may be in the form of either counts or count rates, are some specific fraction of the total radiation emitted from a source. Consequently, it is important to know the factors that affect proportionality, and it is essential to relate measurements either to the total amount of radiation emitted or the activity of the source. To do so requires an understanding of three processes:

Radioactive decay.
Transmission of radiation from a source to a detector.
Interactions of radiation enroute to detector (and as it enters the detector).

All of these processes are statistical in nature; i.e., there exists a random quality similar to that encountered in games of chance (such as rolling dice). In addition, there are sources of error called "systematic," because they appear in the same way in each measurement. Thus, it is necessary to identify and understand such things as:

Sources of random uncertainties or fluctuations (which many textbooks call random error even though they are not errors in the common use of the word).
Sources of systematic error.
Methods of determining magnitudes of these quantities.
Bases for interpreting random uncertainties and treating systematic errors.

These aspects are examined in detail in this module. The factors that have random character are examined first.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define or describe the following terms:
   Radioactive decay.
   Geometry factor (G).
   Interactions.
   Inverse-square law.
   Dead time.
   Average value.
   Standard error.
   Theoretical standard deviation, σ (sigma).
   Standard deviation (s).
   Normal distribution.
2. Explain the variations involved in counting radioactivity by making multiple counts of a single source and

calculating average/mean, standard deviation and probability of obtaining a particular count rate.

3. Count a sample and compensate the reading for background radiation.
4. On a sketch, illustrate the effect of a change in counter geometry on an indicated measurement.

RANDOM PROCESSES

Random processes - associated with source decay, radioactivity transmission and interaction with matter - and the factors involved can be understood by examining a typical arrangement for measuring radiation (Figure 11-1-1). This arrangement consists of a source, a detector and a system for counting detected radiation.

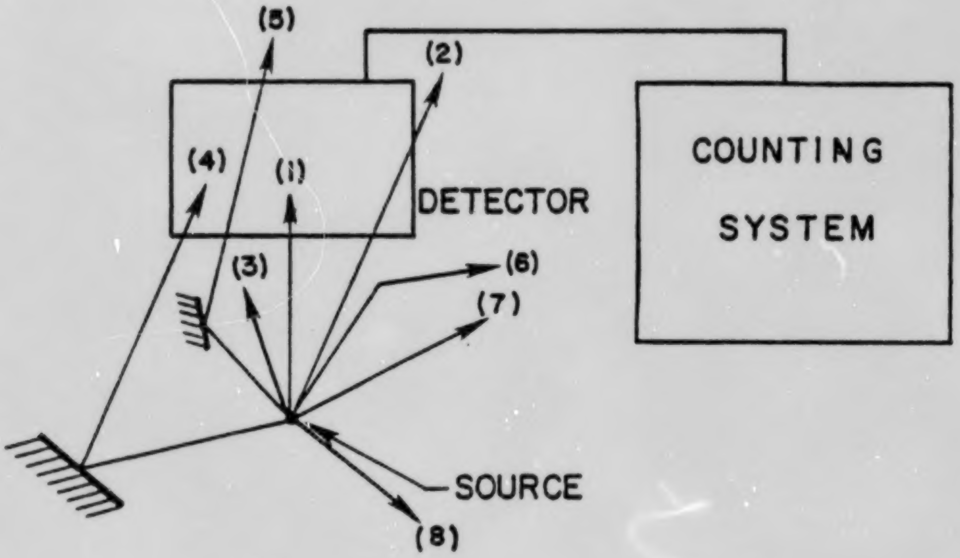

Figure 11-1-1. Typical arrangement for measuring radiation from a radioactive source. (Several processes affecting measurement are shown. See text for discussion.)

The following discussion concentrates on the source and the detector; the counting system will be considered later.

The lines (Figure 11-1-1) represent possible paths of travel for radiation emitted from a thin source material [on a thin foil that does not stop radiation (i.e., no self-absorption)]. The radiation emitted could be beta particles, x-rays or gamma rays. Processes labeled in the figure include:

1. Some radiation enters detector, is absorbed and results in a count recorded by the system.
2. Some radiation enters the detector and passes through with no interaction; no count is recorded.
3. Some radiation, heading toward detector, is absorbed by intervening material (such as air, a cover on the source or any other material between source and detector).
4. Some radiation, heading away from detector, enters detector and is counted after scatter from intervening or surrounding material (such as a table or adjacent surface).
5. Same as (4), except no interaction with detector.
6. Some radiation, which would have entered detector in the absence of any intervening material, is scattered away from detector.

7-8. Some radiation, in the absence of scatter, would never enter the detector.

It is quite clear that only a fraction of the radiation emitted by the source is counted; for the example shown, only two of eight particles emitted by source are counted. The efficiency for counting this specific type emitted particle is 2/8 = 0.25 for this example, which can be expressed in the form of an equation:

$$n \times \varepsilon_t = m$$ Equation 11-1-1
(Equation = Eq.)

where

n = number of particles emitted.
ε_t = total efficiency of counter.
m = number of particles detected by system.

As stated previously, total efficiency (a fraction of the emitted particles counted) is derived as follows:

$$\varepsilon_t = \frac{m}{n} .$$ Eq. 11-1-2

This calculation seems to be relatively simple, except for the random nature of the processes involved. If the same sample is counted several times for exactly the same counting time, the results likely would not be the same.

This variation in counts is caused by the random nature inherent in each of the following processes:

Radioactive Decay:

Decay of radioactive source.

Emission of a particle of a particular type and energy.

Transmission:

Direction of emission.

Interactions of absorption and scatter of radiation emitted.

Detection:

Interactions with detector.

*Resolving time and dead time.

*Sensitivity of electronics in counting system.

The quantities n, m and ε_t (Equation 11-1-1) can be examined from an example series of measurements made on a single sample, each for precisely the same counting time. Assume a radioactive source with a very long half-life (compared with the time for making the measurements - the average decay rate changes by such a small amount that it can be considered constant throughout the measurement period). Also assume some way of knowing n_i, the number of particles emitted by the source during the i-th measurement (that is, during any one of the series of measurements). Results are presented in Table 11-1-1.

This example emphasizes that fluctuations occur for both n (number of particles emitted by source) and m (number of particles detected). Consequently, the value for efficiency fluctuates from measurement to measurement. To report the best value for efficiency, based on this data, summed values of m and n should be used as follows:

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* Random properties of these quantities are coupled with several processes, such as decay and interactions with detector.

TABLE 11-1-1
EXAMPLE OF FLUCTUATIONS IN n, m AND ε_t

i Measurement Number	n_i Number of Particles Emitted During a Fixed Counting Time of t_c Seconds	m_i Number of Particles Counted During t_c Seconds	$(\varepsilon_t)_i$ Fraction Counted (Efficiency)
1	6	2	0.22
2	12	5	0.42
3	8	1	0.125
4	10	3	0.30
5	4	0	0
6	9	4	0.44
7	11	5	0.46
8	7	2	0.29
9	15	6	0.4
10	5	1	0.2
Totals	87	29	

$$\varepsilon_t = \frac{\text{sum of all } m_i}{\text{sum of all } n_i} = \frac{m_1 + m_2 + m_3 + \ldots m_{10}}{n_1 + n_2 + n_3 + \ldots n_{10}}$$

Eq. 11-1-3

$$\varepsilon_t = \frac{\sum_{i=1}^{10} m_i}{\sum_{i=1}^{10} n_i} = \frac{29}{87} = 0.3.$$

In shorthand notation, which is used extensively, efficiency for this case may be derived as follows:

$$\varepsilon_t = \frac{29}{87} = 0.3.$$

An interesting question now arises: Would the same results appear if the set of ten measurements were repeated? The answer is "no." Individual values would vary and the sums would differ; however, if more and more measurements were made, individual measurements, sums and computed efficiencies would all cluster around some value. It is this kind of observation that leads to the conclusion that there is a "true" or "best" value, and that an average value is the best estimate of true value.

The sources of the random fluctuations, what to do to reduce the effect, how to obtain the best value of a set of measurements and what to use as a measure of the uncertainities will be examined in detail. Before this is done, however, the following areas of random processes must be discussed: radioactive decay, transmission and detection.

RADIOACTIVE DECAY

The example in the preceding section demonstrated that the number of particles emitted during a fixed counting time, t_c, varies for each counting period. This fluctuation is caused by random processes that take place inside the nucleus. With minor exceptions, the random nature (or rate of decay) of a radionuclide cannot be controlled. This random characteristic is a fundamental property of radioactive decay, and each different radionuclide decays at different rates. Because of the randomness, it is impossible to predict when any particular radionuclide will decay; however, when a large group of atoms are present, as in any sample, the fraction that will decay within a certain time may be predicted with reasonable certainty.

TRANSMISSION

As noted in the discussion of the processes depicted in Figure 11-1-1, transmission is affected by:

Direction of emission (of radiation).

Processes of absorption and scatter caused by material between source and detector.

Geometry Factor

The geometry factor, G, is a measure of the fraction of radiation that would get to the detector in the absence of any scattering or absorption. This factor is difficult to compute accurately for real sources and detectors; however, there is a relation for the idealized case (a point source of radiation and an infinitesimally thin detector with surface area, S). Also, it must be assumed that the source emits radiation "isotropically," that is, uniformly in all directions (Figure 11-1-2).

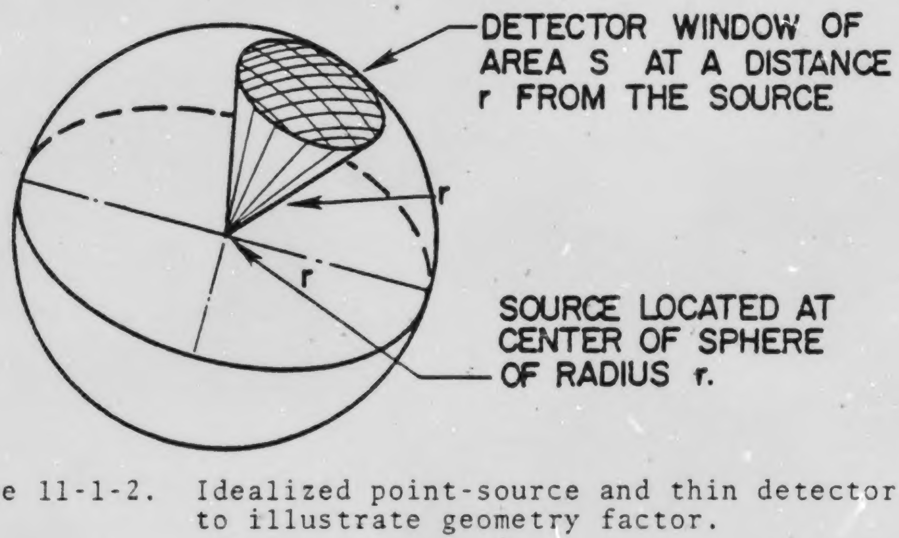

DETECTOR WINDOW OF AREA S AT A DISTANCE r FROM THE SOURCE
r
r
SOURCE LOCATED AT CENTER OF SPHERE OF RADIUS r.

Figure 11-1-2. Idealized point-source and thin detector to illustrate geometry factor.

Since isotropic emission is assumed, the number of particles passing through a unit area of surface can be given by the following equation:

$$\frac{\text{Number of particles emitted in all directions}}{\text{Surface area of a sphere with radius } r} = \frac{n}{4\pi r^2} \text{ particles per unit area} \ldots$$

Eq. 11-1-4

The number of particles entering the detector may then be obtained by multiplying Eq. 11-1-4 by S (area of detector) to obtain:

$$\text{Number of particles entering detector} = n_d = S\left(\frac{n}{4\pi r^2}\right) = n\left(\frac{S}{4\pi r^2}\right) .$$

Eq. 11-1-5

The factor in the last parenthesis is the geometry factor, G - fractional area of window compared with area of entire sphere:

$$G = \frac{S}{4\pi r^2} .$$

Eq. 11-1-6

This relationship illustrates the inverse-square law of attenuation of radiation: amount of radiation entering detector varies inversely as the square of r, the distance from the source to the detector.

This equation applies approximately in practical applications as long as:

Source is small.

Radiation is not scattered.

Distance between source and detector is large (compared with dimensions of detector).

Even though the equation is approximate, it does emphasize two important points. First, total efficiency can be increased by placing source as near as possible to detector. (In fact, it is possible in some applications to place the source inside the detector so that G can be made to approach unity.) Second, intensity of radiation drops rapidly as distance between source and detector is increased. For example, doubling distance reduces radiation entering detector by a factor of four. Tripling distance reduces radiation by a factor of nine and so on.

Interactions

Interaction processes of scatter and absorption are strongly dependent upon type and energy of radiation emitted, and upon properties of any material between source and detector.

The process of absorption for charged particles, such as alpha and beta particles, is different from that for photons, such as x-rays or gamma rays. All charged particles of a particular type and energy are stopped at about the same distance from the source. Some travel a little beyond and some a little short of the average distance or range of travel, but it is possible to prevent all of them from getting to the detector.

In contrast, photons, such as x-rays and gamma rays, penetrate great distances (compared with charged particles of the same energy). There is no average distance in which all photons are stopped. Instead, as learned in Course 1, "Radiation Physics," the number of photons in a beam decreases exponentially (Figure 11-1-3), as demonstrated in Equation 11-1-7:

$$N(x) = N_o e^{-\mu x} \qquad \text{Eq. 11-1-7}$$

where

$N(x)$ = number of photons still in beam after photons have traveled through absorbing slab of material of distance x.

N_o = number of photons incident on surface of absorbing slab of material of thickness x.

μ = absorption coefficient, which depends on material in absorber and energy of photon.

x = distance traveled through slab of material.

The effect of the type of material and thickness of absorber on the value of $N(x)$ is shown in Table 11-1-2 for 1 MeV photons passing through air, aluminum and lead. The thicknesses used are 1 and 2 cm.

From the table, if $N_o = 10^8$ photons of 1 MeV energy, all incident on a slab of aluminum 1 cm thick, 1.93×10^7 photons is transmitted; 2.7×10^4 is transmitted through 1

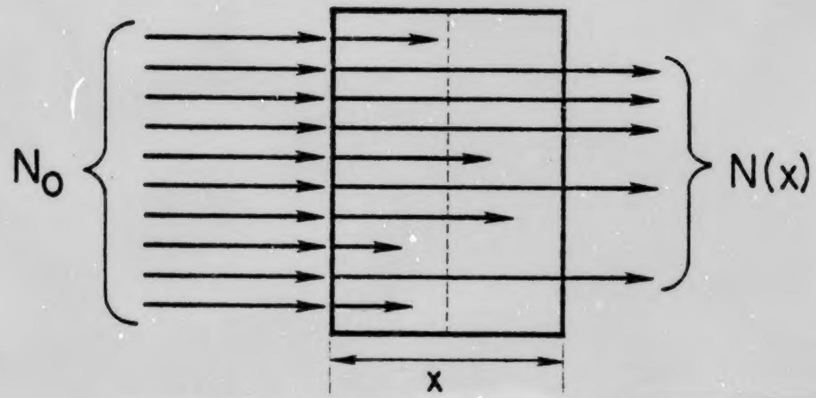


Figure 11-1-3. Representation of exponential attenuation of gamma rays or x-rays.

TABLE 11-1-2

COMPARISONS OF PENETRATION OF 1 MeV PHOTONS THROUGH AIR, ALUMINUM AND LEAD

Material	Density, $\frac{\text{grams}}{\text{cc}}$	For x = 1 cm N/N_0 =	For x = 2 cm N/N_0 =
Air	0.0765	0.99922	0.9984
Aluminum	2.7	0.193	0.037
Lead	11.4	2.7×10^{-4}	7.3×10^{-8}

cm of lead; 9.99922×10^7 is transmitted through 1 cm of air.

If thickness of the absorbers is doubled, the amounts transmitted through air, aluminum and lead, respectively, are 9.984 x 10^7, 3.7 x 10^6 and 7.3. It is obvious that dense, heavy materials with high atomic numbers, such as lead, are better absorbers for photons than light materials, such as aluminum. It is also evident that air absorbs photons of this high energy hardly at all.

DETECTION

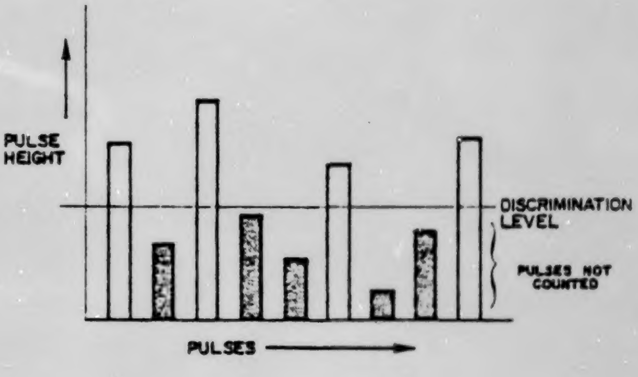
Once radiation enters a detector, it must interact with the detector to produce a signal that can be counted. Too, the detector must be able to respond rapidly so that if two particles enter within a short time interval, both are counted.

This time interval for response is called "resolving time." If the resolving time is 1 μsec (that is 10^{-6} seconds), two particles must be spaced by more than this time interval for both to be counted. The period of time in which the system is unresponsive is called dead time. Resolving time and dead time are discussed further later in this course.

Another factor that can affect the number of particles counted is sensitivity of the counting system. A signal from a detector generally enters a preamplifier, then an amplifier and finally a pulse-shaping circuit before it enters a scaler to be counted. These electronic circuits all have noise pulses, which must be eliminated to prevent the scaler from counting extraneous pulses. Consequently, circuits in an amplifier contain a discriminator level, which can be set just higher than the largest noise pulses (Figure 11-1-4).

If a signal from the detector is smaller than the discriminator level, it is not counted. This concept is important when the detector is a proportional counter, or scintillation counter, either of which produces signals proportional to the energy deposited by a particle in the detector. As a consequence, some of the particles are not counted (even though they enter the detector and interact) because pulses they produce are the same size as the noise pulses the discriminator cuts out.

Figure 11-1-4.
Graph of discriminator level.



The only detail of the interaction process noted here is that, in some cases, proper detector selection can provide detection of each particle that enters. The fraction of entering particles counted is called "intrinsic efficiency," symbolized by ε_i. When $\varepsilon_i = 1$, all particles that enter are counted which frequently is the case for charged particles and low-energy photons.

SUMMARY

Now the processes discussed under radioactive decay, transmission and detection can be pulled together and put into one equation. This equation is an extension of Equation 11-1-1, in that it expands the term ε_t (total efficiency). The number of counts observed when n particles of one particular type are emitted may be calculated as follows:

$$m = n \times G \times f_s \times f_a \times \varepsilon_i \times f_r \times f_d \qquad \text{Eq. 11-1-8}$$

where

m = number of particles counted during time, t_c.
n = number of particles emitted by the course in time, t_c.
G = geometry factor.
f_s = scattering factor.
f_a = absorption factor.
ε_i = intrinsic efficiency.
f_r = resolving time factor.
f_d = sensitivity or discrimination factor.

It is important to note the following:

The number of particles, n, emitted during counting period t_c may be greater than or less than the number of decayed radioactive atoms. Also, the types of radiation emitted may be different. For example, radioactive cobalt, ^{60}Co, emits (with each decay) and ε_i depend on the energy and type of particle. Thus, add to Equation 11-1-8 different terms for f_s, f_a, ε_i, f_d for each type of radiation emitted.

The scattering factor f_s is generally a number greater than 1 and includes scattering from all sources (for example, the mount on which the radioactivity is placed, the walls and floor in the counting chamber, any cover over the source and the air surrounding the source).

The absorption factor f_a is less than 1. It includes absorption by the source itself (if it is thick), any cover over the source, air between the source and the detector, and any window, cover or dead layer in the detector.

The intrinsic efficiency ε_i is 1 or less, depending on material in the detector and the type of radiation. For example, almost all 1 MeV electrons that enter a GM tube are counted; thus, $(\varepsilon_i)_e \cong 1$. But only about 1% of all 1 MeV gammas that enter are counted; thus, $(\varepsilon_i)_{1\ \mathrm{MeV}\ \gamma} = 0.01$.

The geometry factor G is generally less than 1. If the source is thin and it is placed on the surface of the entrance window to the detector, G approaches 0.5. In the absence of scattering and absorption, one-half of all particles emitted would enter the detector. If source is placed far from the detector G becomes very small; if source is placed at the center of the detector, G = 1.

The dead time or resolving time correction f_r can generally be kept close to 1 by keeping the count rate low.

The sensitivity or discrimination-level factor f_d depends on the type of detector and the radiation being measured. It can be made nearly 1 for most cases, and 1 for some cases.

Many of the sources that cause fluctuations, and some of the sources of systematic error, have been identified. The few that remain will be discussed here briefly, and in a subsequent section on counting.

When counting a radioactive sample, the total count includes counts both from natural radioactive sources (in the surroundings) and from cosmic rays. The background radiation counts must be subtracted from observed counts to give a value of m to use in Equation 11-1-8. If m_{sb} represents counts observed (from source plus background), and m_b represents counts from background only, and m_s represents net counts from source alone, and m_{sb} and m_b are measured in separate measurements for the same counting time, t_c, then m_s can be calculated as follows:

$$m_s = m_{sb} - m_b. \qquad \text{Eq. 11-1-9}$$

The value of m_s can now be used in Equation 11-1-8 to solve for n (total number of particles emitted):

$$n = \frac{(m_{sb} - m_b)}{G\, f_s\, f_a\, \varepsilon_i\, f_r\, f_d} = \frac{m_{sb} - m_b}{\varepsilon_t}\,. \qquad \text{Eq. 11-1-10}$$

This equation can be applied if correction factors for each type of radiation (alpha, beta, photons) are known and if total efficiency factor ε_t is known. Since most radioactive sources emit more than one type particle, Equation 11-1-10 must be modified to obtain a more general form. This modification requires different values of f_s, f_a and ε_i for each type particle. Thus, if a radionuclide emits two types of particle (for example, beta particles, β, and gamma rays, γ), the relationship may be shown as follows:

$$n = n_\beta + n_\gamma = \frac{(m_{sb} - m_b)}{G\ f_r\ f_d} \times \left\{\left(\frac{1}{f_s\ f_a\ \varepsilon_i}\right)_\beta + \left(\frac{1}{f_s\ f_a\ \varepsilon_i}\right)_\gamma\right\}.$$

Eq. 11-1-11

If more than two types of radiation are emitted, simply add additional terms for each different type.

The total number of particles, n, emitted can be related to activity, A, if how the radionuclide decays is known (i.e., how many of each type of particle is emitted with each nuclear transformation). This added factor makes the calculation of activity quite complicated because the decay of many radionuclides is complicated.

The general idea can be learned, however, by using a simple case, such as the decay of carbon-14 (^{14}C). This radionuclide emits just one beta particle for each nuclear transformation, and its half-life (∿ 5000 years) is long compared with counting time t_c. Consequently, no correction is needed for radioactive decay. If t_c is expressed in seconds, activity A is in units of nuclear transformation per second (ntps):

$$A(\text{ntps}) = \frac{n}{t_c\ (\text{sec})} = \frac{(m_{sb} - m_b)/t_c\ (\text{sec})}{G \times f_s \times f_a \times \varepsilon_i \times f_r \times f_d} .$$

Eq. 11-1-12

EXAMPLE

Given:
A sample of ^{14}C counted for a time (t_c) of 1000 seconds; a total count (m_{sb}) of 500 counts is obtained; background is also counted for 1000 seconds (with ^{14}C sample completely removed from vicinity of counter). The result (m_b) obtained is 100 counts.

Also,

$G = 0.5$ (Half the β particles enter detector in the absence of scatter and absorption.)

$f_s = 1.2$ (20% of the particles that were heading away from detector are scattered back into detector.)

$f_a = 0.3$ (70% of particles are absorbed before they enter detector.)

$\varepsilon_i = 1$ (All beta particles that enter detector interact to give a signal.)

$f_r = 1$ (Count rate is low so that no two particles enter during resolving time.)

$f_d = 0.9$ (Discriminator cuts out 10% of signals from detector.)

Find: Overall efficiency.

Solution:
Substitute these quantities into righthand side of Equation 11-1-12:

$$A(\text{ntps}) = \frac{(500 - 100)/1000 \text{ seconds}}{0.5 \times 1.2 \times 0.3 \times 1 \times 1 \times 0.9}$$

$$A(\text{ntps}) = 2.47 \text{ nuclear transformations per second.}$$

The total number of β particles emitted during counting period (t_c = 1000 seconds) is as follows because one beta particle is emitted per nuclear transformation of ^{14}C:

$$n = 1000 \text{ sec} \times \frac{2.47}{\text{sec}} = 2470 \text{ beta particles.}$$

Also, overall efficiency ε_t for this example may be derived:

$$\varepsilon_t = G \times f_s \times f_a \times \varepsilon_i \times f_r \times f_d = 0.162.$$

This example clearly indicates multiple causes for disagreement between observed counts, m_{sb}, and actual number of particles emitted. It is simple to correct for background count, but other factors are much more difficult to determine. Errors introduced by use of incorrect values for these quantities are the systematic errors mentioned in the introduction. These errors will be discussed in more detail in later modules in this course. Most will be determined experimentally.

STATISTICAL ANALYSIS OF RANDOM FLUCTUATIONS IN COUNTING DATA

Now that most of the important sources of systematic errors and random fluctuations have been identified, methods of specifying magnitude of random "errors" can be discussed. Two frequently misused quantities, which relate to the effects of systematic errors and random errors, are accuracy and precision.

Accuracy is a measure of deviation of an average value of measurements from a "true" value. In other words, it is assumed that there exists a true mean symbolized by μ, for which $\bar{m}$, the average of the data, is the best estimate. Random processes do cause the average to deviate from this true mean value. Consequently, random fluctuations have been called random errors.

(This terminology will be used for simplicity and consistency with other reference sources; however, the distinction between the two quantities - systematic error and random error - should be kept in mind.)

It is better to measure random errors by a quantity called "precision," a measure of how much the data fluctuates, through range of values below and above the average value. "High precision" means there is very little spread in the values; "low precision" means the spread is large.

The precision of data may be measured by a quantity called standard deviation, for which the symbol s or σ (sigma) may be used, depending on whether the standard deviation is computed from a set of measurements or from statistical theory. Standard deviation (s) is a best estimate of theoretical value, σ, just as $\bar{m}$ is a best estimate of true value, μ.

Once a best estimate, $\bar{m}$ is known, the next step is to compute (from statistical theory) a measure of its precision. This precision index is called standard error of the average (to distinguish it from "standard deviation for the data"). It is represented by the symbol $\sigma_{\overline{m}}$.

The measure of accuracy is more difficult to obtain directly than is the measure of precision because it requires knowledge of the true mean. Elimination of all systematic errors, or evaluations of their magnitude, must be achieved to correct the average value measured to a true mean.

Because evaluations of systematic errors (discussed in previous section) are difficult, calibrated standards are employed to obtain a best value for total efficiency. To do the best work, a standard is required for each radionuclide being measured.

COMPUTATION OF AVERAGE AND STANDARD DEVIATION

The method for computing average and standard deviation may be demonstrated by using the set of data tabulated in Table 11-1-3. These data consist of 20 measurements extracted from the set of 400 measurements plotted in Figure 11-1-5.

The first column contains a tabulating index, i, which simply keeps track of the data. The second column contains the measured counts, m_i, each obtained in a counting time of $t_c = 20$ seconds.

The third column contains deviation of each measurement from the average, $\overline{m} - m_i$. The fourth column contains the square of each deviation, $(\overline{m}-m_i)^2$.

The totals of the various columns appear at the bottom of the table.

TABLE 11-1-3

COUNTING DATA OF SAMPLE PLUS BACKGROUND

Index i	Observed Count m_i	Deviation from Average $(\bar{m} - m_i)$	Square of Deviation $(\bar{m} - m_i)^2$
1	10123	63.85	4076.82
2	10127	59.85	3582.02
3	10234	-47.15	2223.12
4	10246	-59.15	3498.72
5	10284	-97.15	9438.12
6	9987	199.85	39940.02
7	10128	58.85	3463.32
8	10282	-95.15	9053.52
9	10112	74.85	5602.52
10	10051	135.85	18455.22
11	10268	-81.15	6585.32
12	10254	-67.15	4609.12
13	10310	-123.15	15165.92
14	10209	-22.15	490.62
15	10127	63.85	4076.82
16	9950	236.85	56097.22
17	10311	124.15	15413.22
18	10385	198.15	39263.42
19	10215	-28.15	792.42
20	10134	52.85	2793.12
Totals	203737	0.00	244620.60

Average $\bar{m}$ is a best estimate of the true mean, μ, obtained by summing all data m_i and dividing by total number of measurements N, as follows:

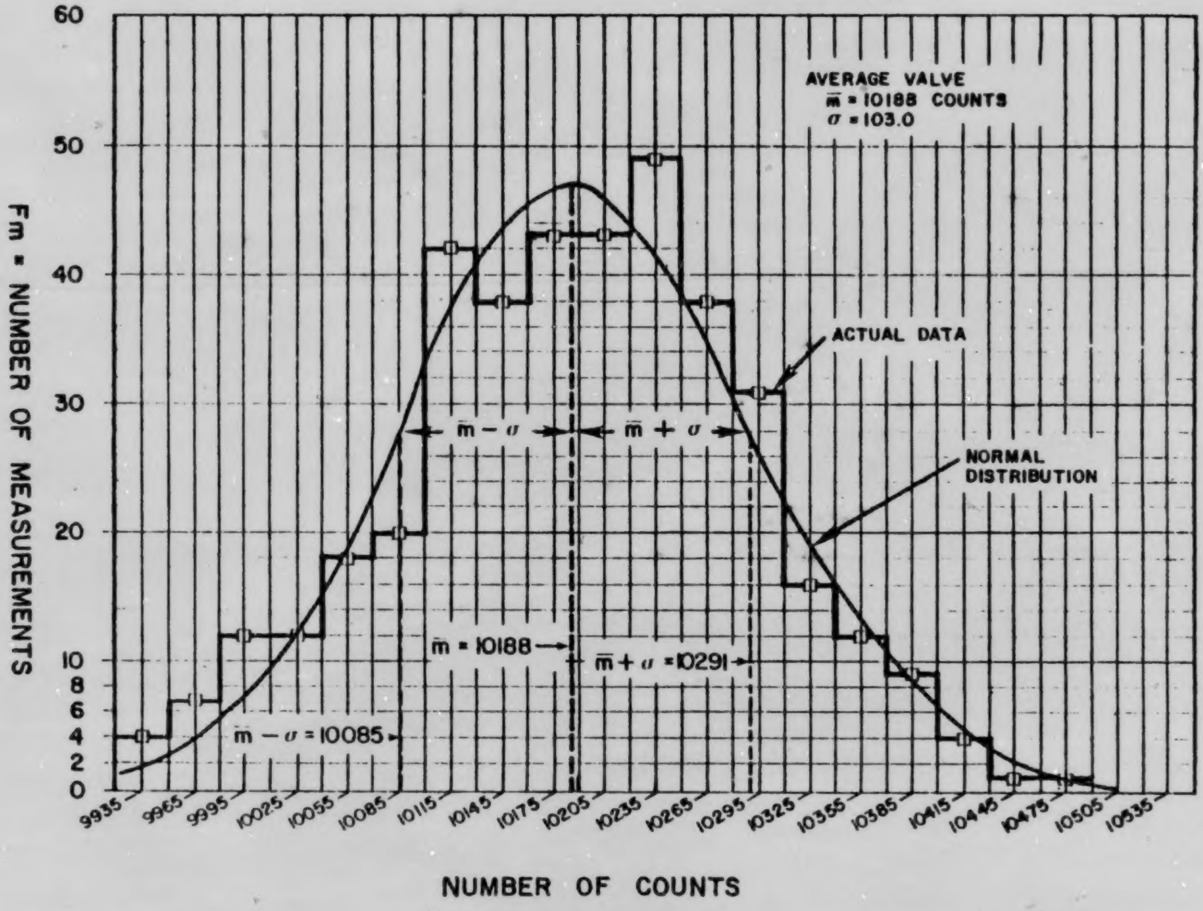


Figure 11-1-5. Distribution of 400 measurements compared with normal distribution. (Counting interval = 30 counts.)

$$\bar{n}_{sb} = \frac{\sum_{i=1}^{N} m_i}{N} \qquad \text{Eq. 11-1-12}$$

$$= \frac{203737}{20} = 10186.85.$$

Subscript sb indicates that count includes source-plus-background radiation.

Standard deviation obtained from a set of data is given as s. This quantity is a best estimate of theoretical standard deviation σ. It is obtained from the square-root of the sum of the squares of the deviations of such measurement from the average divided by N - 1:

$$s = \left(\frac{\sum_{i=1}^{N} (\bar{m} - m_i)^2}{N-1} \right)^{\frac{1}{2}} \qquad \text{Eq. 11-1-13}$$

$$= \left(\frac{244620.6}{19} \right)^{\frac{1}{2}} = 113.5 \simeq 114.$$

About 68% of the 20 values (0.68 x 20 = 13 to 14) have deviations from the average of less than $\sigma = 114$: i.e., 6 or 7 can be expected to deviate from the average by more than 114. An examination of the third column shows that 6 values have deviations greater than 114, which is in good agreement with the statistical theory.

ESTIMATION OF THE STANDARD DEVIATION

Statistical theory predicts that the standard deviation for counting data is given (approximately) by the square-root of true mean μ. Since average $\bar{m}$ is a best estimate of true mean, a best estimate of the standard deviation is derived as follows:

$$\sigma \approx \sqrt{\mu} \approx \sqrt{\bar{m}}. \qquad \text{Eq. 11-1-14}$$

Sigma computed this way can be compared with the s obtained for the set of data in Table 11-1-3 in which

$$\sigma \approx \sqrt{10187} \approx 102.$$

This quantity does agree with the statistical expectations for s obtained from 20 measurements. Also, this value of σ compares more favorably with the value of s as the number of measurements is increased.

Such a method of estimating standard deviation may be used whenever there are a few (1 to 3) measurements because the computation of s (Eq. 11-1-13) is not accurate in this case. Experience has shown that the estimate given by Equation 11-1-14 is adequate for such practical applications.

ESTIMATING THE STANDARD ERROR OF THE AVERAGE

Up to this point, sufficiently large quantities of individual measurements have not been introduced so that they could be examined for the spread of individual measurements around average value $\bar{m}$. To know how a set of values of $\bar{m}$ would spread around true mean, μ, it is necessary to take many sets of data and examine the spread of each value of $\bar{m}$ around the average obtained from using all the data.

The results of such a study would show that the standard error of average $\sigma_{\bar{m}}$ is related to standard deviation σ as follows:

$$\sigma_{\bar{m}} = \frac{\sigma}{\sqrt{N}} . \qquad \text{Eq. 11-1-15}$$

Either s or the theoretical value of σ could be used in this equation. For example, using s from the previous example gives this determination:

$$\sigma_{\bar{m}} \simeq \frac{s}{\sqrt{N}} = \frac{113.5}{\sqrt{20}} = 25.4 .$$

But, $\sigma = \sqrt{\bar{m}}$ gives this determination:

$$\sigma_{\bar{m}} = \frac{\sigma}{\sqrt{N}} = \frac{\sqrt{\bar{m}}}{\sqrt{N}} = \frac{102}{\sqrt{20}} = 22.8 .$$

Obviously, these two values for the standard error of the mean are in close agreement, and either method can be used to make a satisfactory determination.

REPORTING OF $\bar{m}$ AND THE MEANING OF THE STANDARD ERROR

The value for average and the statistical uncertainty associated with it can be reported as follows:

$$\bar{m} \pm \Delta\bar{m} = \bar{m} \pm k\sigma_{\bar{m}} \qquad \text{Eq. 11-1-16}$$

where k can be any value, and the precision index $\Delta\bar{m} = k\sigma_{\bar{m}}$ can be interpreted in a manner somewhat similar to that presented for $k\sigma$ (for example, if $k = 1$, statistical theory predicts that there are about 68 chances out of 100 that true mean μ is included in the interval $\bar{m} \pm \sigma_{\bar{m}}$).

Use the preceding equation with data from Table 11-1-3 to obtain:

$$\bar{m}_{sb} \pm \sigma_{\bar{m}_{sb}} = 10186.85 \pm 22.8 \ .$$

Since the standard error of the average shows that $\bar{m}$ is uncertain in the last two digits in front of the decimal point, it makes no real sense to retain the decimals either in $\bar{m}_{sb}$ or $\sigma_{\bar{m}_{sb}}$.

> The average value of the gross count from the source, plus background obtained from 20 measurements, is presented here with a standard error of 1 $\sigma_{\bar{m}}$:
>
> $$\bar{m}_{sb} \pm \sigma_{\bar{m}_{sb}} = (10187 \pm 23) \text{ counts per seconds.}$$

Descriptive words, such as those used in the preceding example, should always be included when reporting results. If these descriptive words are not included, the student has no way of knowing what the quoted uncertainties mean. Results can be presented as count rate per second, as follows:

$$\bar{r}_{sb} \pm \sigma_{\bar{r}_{sb}} = \frac{\bar{m}_{sb} \pm \sigma_{\bar{m}_{sb}}}{t_c} \qquad \text{Eq. 11-1-17}$$

$$= \frac{10137 \pm 23}{20 \text{ sec}}$$

$$\bar{r}_{sb} \pm \sigma_{\bar{r}_{sb}} = 509.4 \pm 1.2 \text{ counts/sec.}$$

From the standard error, both the units and first decimal are uncertain in the value of $\bar{r}$. It is good practice to retain at least two uncertain figures in the result; in fact, it is essential to do this to prevent additional errors such as result from rounding off numbers.

CORRECTING FOR BACKGROUND AND PROPAGATION OF ERRORS

The simplest example of correction for background is obtained when both the gross count and the background are counted for the same time interval. Take as an example a background count, $\bar{m}_b \pm \sigma_{\overline{mb}}$, obtained for $t_c = 20$ sec:

$$\bar{m}_b \pm \sigma_{\bar{m}_b} = 26.1 \pm 1.2 \text{ counts per 20 sec.}$$

Net count and uncertainty are as follows:

$$\bar{m}_s = \bar{m}_{sb} - \bar{m}_b \qquad \text{Eq. 11-1-18}$$

$$\sigma_{\bar{m}_s} = \sqrt{\sigma_{\bar{m}_{sb}}^2 + \sigma_{\bar{m}_b}^2} .$$

Substituting (to complete the example):

$$\overline{m}_s = \overline{m}_{sb} - \overline{m}_b = 10187 - 26 = 10161$$

$$\sigma_{\overline{m}_s} = \sqrt{23^2 + 1.2^2}$$

$$\approx 23.$$

For this example, the standard error for the background count is so much smaller than the standard error for the gross counts of source-plus-background, that the standard error of the net count is not changed enough to include it in the final value. Thus, the value for net count from source may be reported as:

$$\overline{m}_s \pm \sigma_{\overline{m}_s} = 10161 \pm 23 \text{ counts per 20 seconds}$$

or

$$\overline{r}_s \pm \sigma_{\overline{r}} = 508.1 \pm 1.2 \text{ counts/sec.}$$

In many instances, sample-plus-background and background may be counted for different times. In fact, the length of time selected for counting the background depends to a large extent upon the relative magnitude of count rates of source and background.

If results are like those of the example (above), the background need not be counted for as long as the sample. If, however, the sample count rate approaches that of the background, then the background should be counted for a longer time to reduce the standard error in count rate.

DISTRIBUTION OF DATA

The statistical analysis of data affected by random processes requires study of very large numbers of measurements. Such study has been done with theoretical models based on behavior of large populations, and by comparisons of experimental results with predictions of a particular theoretical model. The models used to describe fluctuations in counting data are binomial distributions, poisson distributions and normal distributions. It is beyond the scope of this module to deal with these in detail, or to discuss their applicability and limitations. For the purpose of this study, it is adequate to use normal distribution as a model to describe the spread of values observed in counting data, and to interpret standard deviation.

The nature of distribution of data and a comparison with theoretical models (Figure 11-1-6) may be derived from the plot-of-distribution of 37,075 actual measurements for which average value $\overline{m} = 98.06$ counts. Experimental data are presented by the square symbols (⊡) and a theoretical curve (based on normal distribution) is represented by a solid line.

The values plotted on the vertical axis are the number of measurements observed; values of counts are plotted on the horizontal axis. As an example, 1471 measurements have a value of 100 counts.

Theoretical prediction of the number of measurements that should turn up with 100 counts is obtained from the normal distribution equation, which gives the probability that, for a particular value given, true mean $\mu \approx \overline{m} = 98.06$, and standard deviation $s \approx \sigma = \sqrt{\overline{m}} = 9.902$. Substitute these values into the normal distribution equation:

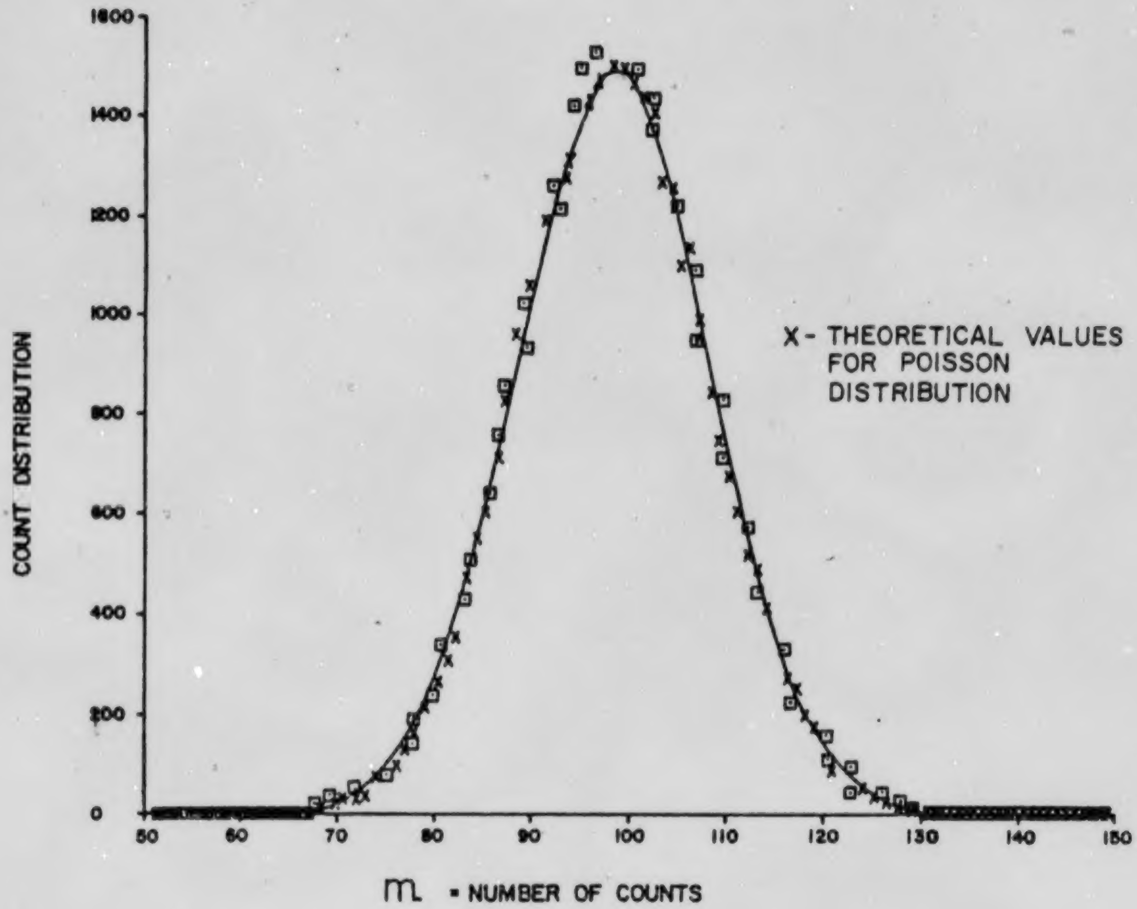


Figure 11-1-6. Comparison of experimental counting data with prediction of the normal distribution.

$$P(n) = \frac{1}{\sqrt{2\pi\sigma}} e^{-(\mu-n)^2/2\sigma^2}$$

Eq. 11-1-19

$$P(100) \simeq \frac{1}{\sqrt{2\pi} \times 9.902} e^{-(98.06-100)^2/2 \times 9.902^2} = 0.0395.$$

This value, P(100), is the probability that a measurement having 100 counts will be observed. In other words, P(100) is the fraction of the total measurements that theory predicts will come up with 100 counts. Thus, 0.0395 x 37,075 = 1464 measurements are expected, which compares favorably with the 1471 observed.

In principle, the experimental and the predicted values agree only when the number of measurements made are very large.

To show the effect of reducing a population of measurements, the plot of the distribution of 400 measurements is shown (Figure 11-1-5). The number of measurements is so small that several measurements must be grouped together to plot their distribution. The values plotted are the number measurements in intervals of 30 counts; i.e., 9920 to 9949, 9950 to 9979, and so on, plotted at midpoints, 9935, 9965, and so on. Thus, there are four measurements with values between 9920 and 9949, seven measurements with values between 9950 and 9979, and so on.

Because the number plotted at the midpoints of these intervals is actually the number in the interval, the plot is in the form of rectangular bars, called "histograms." Each bar contains F_m measurements distributed over 30 units of counts.

If all values of F_m that are plotted are added, these would equal the total number of measurements, which is 400.

The jagged character of the distribution of experimental data, compared with theory, is a direct consequence of the random nature of the data and the small sample population of 400 in contract with the 37,075 shown in Figure 11-1-5.

The fact that the experimental data approaches the theoretical model, in the limiting case of large population, supports the idea that normal distribution can be used to interpret the meaning of standard deviation. This deviation is shown on the plot as the limits of a zone between $\overline{m} - \sigma$ below average and $\overline{m} + \sigma$ above average.

If the area enclosed by this zone is computed from the theoretical relation for normal distribution, it is ∿ 0.68 of the total area under the normal distribution curve. This statement is interpreted to mean that 68% of the 400 measurements are expected to have values that lie within the zone, $\overline{m} \pm \sigma = 10188 \pm 103$. Because the total number of measurements is 400, then $400 \times 0.68 = 272$ of the measurements are expected to be within $\overline{m} \pm \sigma$. Also, the number of measurements expected to lie outside this zone could be stated to be $0.32 \times 400 = 128$. That is, 128 measurements would have values that deviate from the average by more than 103.

Bounds placed on the zone around the average can be made as large or as small as desired. The extent of this zone controls the fraction of the data expected to be found encompassed by the limits. Thus, if the zone is made $\overline{m} \pm 2\sigma$, it will extend from $10188 - 206$ to $10188 + 206 = 9982$ to 10394. From normal distribution, this zone contains about 95% of the data; i.e., only 5%, or $0.05 \times 400 = 20$ measurements, are expected to lie outside the zone.

Various values for the probabilities of observing measurements within limits $\overline{m} \pm k\sigma$ and outside these limits are presented in Table 11-1-4.

TABLE 11-1-4

PROBABILITIES OF OBSERVING MEASUREMENTS WHICH DEVIATE FROM THE MEAN BY MORE AND LESS THAN THE DEVIATION SHOWN

± Deviation	Possibilities of Observing Measurements	
	Less Than Deviation	More Than Deviation
0.6745σ	0.500	0.500
1.0σ	0.6826	0.3174
2.0σ	0.9546	0.0454
2.5σ	0.9876	0.0124
3.0σ	0.9973	0.0053
4.0σ	0.99937	0.3×10^{-5}
5.0σ	0.99999962	3.8×10^{-7}

Results in the table show that the larger the deviation, the greater the probability that a measurement will be within the deviation from the mean. For example, if the deviation selected is 3σ, then the interpretation is that 99.73% of the data will deviate from the mean by less than 3σ. For our example, this means that $\sim$ 399 of the 400 measurements will lie between $\overline{m} \pm 3\sigma = 10188 \pm 309$, or that $\sim$ 1 will lie outside these bounds. The predicted and observed results for this set of 400 measurements are presented in Table 11-1-5. Even with the small population of 400 measurements, agreement is remarkably good.

TABLE 11-1-5

COMPARISON OF OBSERVED AND PREDICTED RESULTS FOR THE 400 MEASUREMENTS PLOTTED IN FIGURE 11-1-5

	Number of Measurements With the Deviation	
Deviation	Predicted	Observed
0.6745σ	200	198
1.0σ	273	273
2.0σ	382	384
2.5σ	395	396
3.0σ	399	400

In summary, standard deviation of the data (or some multiple of it) can be used to measure spread of the data. When coupled with normal distribution, the predicted and observed spread are in reasonably close agreement - enough so to justify use of normal distribution for estimation of probabilities.

SAMPLE CALCULATIONS FOR PRACTICAL EXPERIMENT

In the preceding examples, a large number of measurements were made to show the nature of statistical fluctuations and their interpretation. For a more realistic case, several measurements of the sample may be made for longer counting times. The only reason for making several measurements is to keep a running check on the operation of the equipment. Actually one measurement for 10 minutes is as good as 10 measurements each, for 1 minute. Consider the following example.

EXAMPLE

Given:

Experimental data for 10 measurements of a radioactive sample-plus-background. Counting time for each measurement is 1 minute. Shown off to the side, and denoted by m_{10}, is an additional measurement taken for 10 minutes.

i		m_i (counts in 1 minute)	
1		29	
2		36	
3		19	
4		26	
5		24	m_{10} = 282 counts/ 10 min
6		37	
7		35	
8		29	
9		30	
10		34	
	$\bar{m} = \sum_{1}^{N} m_i$	299	

Find:

Average counts/min.

Solution:

$$\bar{m} = \frac{\sum_{i=1}^{N} m_i}{N} = \frac{299}{10} = 29.9 \text{ counts/min.}$$

(Subscript sb omitted for convenience.)

1. Analysis of the set of 10 individual measurements:

$$\sigma_{\bar{m}} = \frac{\sigma}{\sqrt{N}} \approx \frac{\sqrt{\bar{m}}}{\sqrt{N}} = \frac{\sqrt{29.9}}{\sqrt{10}} = 1.73 \text{ counts/min}$$

$$\bar{m} \pm \sigma_{\bar{m}} = 29.9 \pm 1.7 \text{ counts/min.}$$

2. Analysis of the single measurement for 10 minutes:

$$m_{10} \pm \sigma_{m_{10}} \approx m_{10} \pm \sqrt{m_{10}}$$

$$282 \pm \sqrt{285} = 282 \pm 16.9 \text{ counts/10 min}$$

or

$$28.5 \pm 1.7 \text{ counts/min.}$$

(Note: Assume that $\sigma \approx \sqrt{m}$ when there is one measurement.) Standard error is the same for the two values expressed as for the counts per minute. Also, the two values agree within limits set by the standard error. As a further example, the effect of increasing time can be seen; single measurement of this same sample for 30 minutes resulted in 845 counts. The rate for 1 minute with the standard error is

$$\frac{m_{30} \pm \sigma_{m_{30}}}{30} = \frac{845 + \sqrt{845}}{30} = 28.2 \pm 1 \frac{\text{counts}}{\text{min}} .$$

It is evident that the factor-of-3 increase in time has reduced the standard error by a factor of $\sqrt{3}$. There is a trade-off between error and available counting time.

The examples examined suggest that, in practice, several measurements are adequate and that the larger the total count accumulated, the smaller the random error in the count rate becomes. It also shows that standard deviation for a single measurement, or of the total of several measurements, can be estimated by use of the relation, $\sigma = \sqrt{m}$.

EXERCISES

1. Define the following factors and give typical values:

   G = geometry factor.
   f_s = scattering factor.
   f_a = absorption factor.
   ε_i = intrinsic efficiency.
   f_r = resolving time factor.
   f_d = sensitivity factor (or discrimination factor).

2. The following one-minute counts were taken on a sample, including background:

   830
   790
   824
   736
   801
   826
   799
   782
   795
   829
   841
   811
   803
   840
   823
   819
   826
   788
   798
   800
   814
   819
   803
   795
   762
   785
   852
   843
   787
   869

   Find the mean $\overline{m}$, sample standard deviation s, and standard error $\sigma_{\overline{m}}$.

How many counts are within ± 1σ, ± 2σ? Assume $\sigma \approx s$.
How many counts would you expect to be within these limits if count is random?
Compute theoretical standard deviation σ; compare with best estimate s.
Divide data into intervals of 10 counts each (i.e., 800-809, 810-819, etc.); plot data; also plot normal distribution equation (Eq. 11-1-11). Explain why observed data does not fall on curve.

3. A 30-minute count, using the conditions given above, produces a count rate of 798 counts/minute.

   Find standard deviation of this count.
   Which is more accurate, the 30 one-minute counts (in Exercise 2) or this one 30-minute count? Explain.

4. Explain any difference between accuracy and precision; use hypothetical data that you chose.

5. The following background data was collected by using one-minute counting times:

42	14	43	37	44
40	15	39	54	36
39	11	17	15	
26	31	51	29	
10	25	9	39	
25	19	15	30	

   Find mean, standard deviation for the data, and standard error for mean. What is the probability of finding a count of exactly 50?
   What is the approximate probability of finding a count of greater than 50?
   What would you expect if you had successive counts of 54 and 59?

6. A one-minute count of a ^{60}Co source-plus-background was 329 cpm. Background alone was 87 cpm. Compute value and standard deviation for source alone.

7. In the previous exercise, background count was 473 for a five-minute count.
What is value and standard deviation for source alone? Has the longer background count lowered error and raised accuracy? Explain.

PROCEDURES

None,

DATA TABLE

None.

DATA TABLE

None.

REFERENCE MATERIALS

Chase, G.D. and Rabinowitz, J.L. Principles of Radioisotope Methodology. Minneapolis, MN: Burgess Publishing Co., 1965.

Frankel, Robert. Radiation Protection for Radiologic Technologists. New York: McGraw-Hill, 1976.

Hall, Eric J. Radiation and Life. New York: Pergamon Press, 1976.

Ouseph, P.J. Introduction to Nuclear Radiation Detectors. New York: Plenum Press, 1975.

Price, William James. Nuclear Radiation Detection. 2nd ed. New York: McGraw-Hill, 1974.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space. (Note: a slide rule, calculator, or square root chart is needed.)

1. Given a typical setup of a source, not inside the detector, and system for counting radiation, briefly describe the possible paths of travel for radiation emitted from a thin source material with no self-absorption. The radiation could be beta particles, x or gamma rays. (Your response should concentrate on the source and the detector.)

2. What is radioactive decay? Why does radioactive decay affect the variation in counts obtained when counting a radioactive source?

3. What is the geometry factor? What is the relationship between the geometry factor and the number of counts obtained when counting a radioactive source?

4. What implication does inverse square law have for positioning the detector and source for maximum efficiency?

5. Describe the change in intensity of radiation if the detector moved from within one inch of the source to two inches away from a point source.

6. How do the type and energy of the radiation emitted by the source affect the interaction processes of absorption?

7. Any material between the source and the detector affects absorption. Compare air, aluminum and lead as absorbers for photons.

8. Briefly describe how dead time and discriminator level affect radiation counting.

9. Describe the normal distribution curve. Explain why the normal distribution can be useful for experimental counting data.

10. Differentiate between μ and $\overline{m}$.

11. Define standard deviation. Differentiate between σ and s.

12. Define standard error of the mean ($\sigma\overline{m}$). What purpose does this statistic serve? What is the relationship between standard error of the mean and standard deviation?

13. Ten counts of a single source plus background were made. Given the following data, calculate mean, standard deviation and standard error of the mean.

INDEX	OBSERVED COUNTS
1	1,000
2	950
3	1,050
4	1,040
5	1,030
6	900
7	1,005
8	995
9	1,060
10	1,010

14. Write (report) the results obtained for Question 13 such that the results can be clearly interpreted. (Let K = 1)

15. In Question 14, how many counts are within $\pm 1\sigma$? $\pm 2\sigma$? Assuming $\sigma = s$, how many counts would you expect to be within these limits?

16. If instead, of taking ten one-minute counts of a sample (as in Question 14), you took ten ten-minute counts of the sample, would you expect a difference in the standard error of the mean? Why?

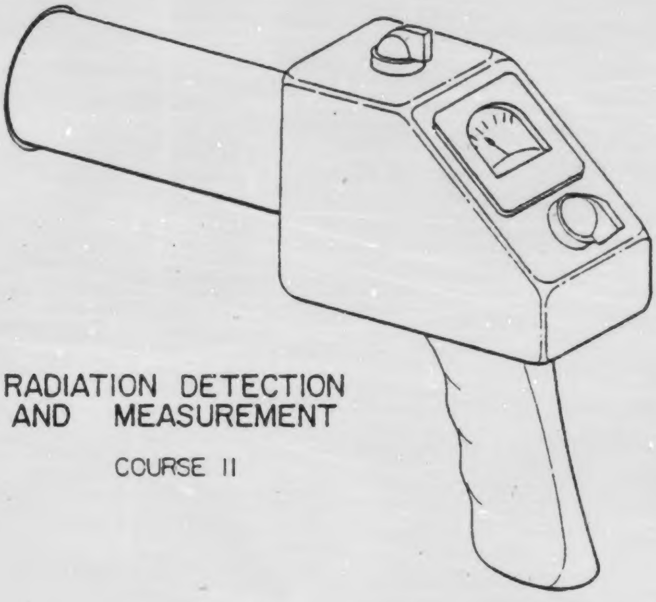
17. Given $\mu \approx \bar{m} = 95$ and $s \approx \sigma = \sqrt{\bar{m}} = 10$, how many measurements out of 35,000 would be expected to give a count of 100?

18. Compensate for background given a mean value for source-plus-background of 1000± 20 CPS and mean value for background alone is 260± 8 CPS.

19. Given the equation for the geometry factor $G = \frac{5}{4\pi r^2}$, describe the relationship between detector size and indicated count rate.

NUCLEAR TECHNOLOGY

MODULE 11-2 GAS-FILLED DETECTORS I: IONIZATION DETECTORS.



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

	MODULE 11-1	Statistics of Radiation Counting
⇨	MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
	MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
	MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
	MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
	MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
	MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
	MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

Since nuclear radiations cannot be directly detected by the human senses, they must be detected and measured by other means. One of the earliest and best methods of detection took advantage of the fact that charged particles or ions are produced by an interaction of nuclear radiation with matter. One such instrument developed on this principle - the ionization chamber - detects and measures radiation by collecting the ionization (electrical charge) produced within it by the passage of radiation.

Ionization chamber instruments fall into two broad categories: "integrating" and "nonintegrating." Integrating means combining or blending together; thus an integrating counter combines counts into an average value, whereas a nonintegrating counter detects each pulse separately.

Integrating ionization chambers are further subdivided into two types. The first collects the total charge (or dose) produced by radiation in a given or set time period; this type is called a dose meter.

The other type is known as a *mean-level detector* because it averages the current produced by radiation over very short periods of time - generally fractions of a second - and provides a measure of the rate of radiation. This type is called a rate or rate-level meter.

Nonintegrating ionization chambers detect each ionizing occurrence as an individual event or pulse; for this reason, they are known as event or *pulse counters*.

Ionization chambers of the nonintegrating type are used in laboratory counting systems to determine radioactivity disintegrations. The integrating ones are used for personnel dose meters. The integrating rate-level type are widely used as portable radiation monitoring instruments in area surveillance.

In this module, basic principles of ionization chambers are discussed, along with many applications of these useful and versatile devices.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Differentiate between and briefly describe characteristics of integrating, nonintegrating and mean-level ionization detectors.
2. Calculate the current produced by a given exposure in an ionization chamber of known size.
3. Sketch a simple ionization chamber circuit and label all its parts.
4. Draw a saturation curve for an ionization chamber.
5. Show, with the aid of a graph, how ionization chambers

can be used to differentiate between various kinds of radiations.

6. Describe basic principles of ionization chamber applications; give two examples of each.
7. Sketch and label all parts of a simple pocket dosimeter.
8. Define the following terms in terms of their significance to ionization detection:
    - Mean-level detector.
    - Pulse counter.
    - Ion pair.
    - Recombination.
    - Saturation voltage.
    - Saturation current.
    - W.
    - Discriminator.
    - Air equivalence.
    - Energy dependence.
    - Spectroscopy.
    - Efficiency.

IONIZATION DETECTORS

When ionizing radiation passes through matter, it transfers some of its energy to the medium through which it is traveling. If the medium is a gas, energy given up by the radiation will generally result in removal of orbital electrons (i.e., ionization) from the neutral gas atoms. In this manner, positive gas ions and free electrons are formed; these are known as ion pairs.

On the average, it takes about 34 eV of energy to form one ion pair in a gas; about half of this energy actually goes to production of the ion pair, while the remainder merely excites or raises the energy level of the gas atoms.

The number of ion pairs produced in a given volume of gas depends upon many factors, including the amount of energy associated with the nuclear particle or radiation and the type of particle.

Alpha particles, being relatively heavy and slow moving on the atomic scale, have a very short range - usually a few centimeters at most. Because they move relatively slowly, alpha particles spend more time in the neighborhood of individual atoms as they travel through the gas, and therefore have a greater chance of producing ionization. Alpha particles, therefore, produce a large number of ion pairs over very short paths. Such particles are said to have a high specific ionization, meaning that they produce many pairs in a short distance; in air, an alpha particle may produce tens of thousands of ion pairs along each centimeter of its path.

Because of their generally higher velocities and smaller charges, beta particles (electrons) leave fewer ion pairs behind as they travel through a gas - perhaps a few hundred per centimeter. They also have a longer range before interaction.

Gamma rays and x-rays, which are uncharged and interact by indirect collisional means, may produce only a few ion pairs per centimeter of track and, in general, travel great distances before suffering an ionizing collision.

An immediate difference can be noted. In a small or finite sized ion chamber, generally all of a particle's energy is expended, and many ion pairs are produced. In contrast, only a small amount of energy is expended by a gamma ray. This difference has a practical value, which is discussed later in this module.

Ion pairs produced in gas can be collected and measured in a relatively simple fashion. If a pair of electrodes - one positively charged and the other negatively charged - is placed into the gas, ions are attracted toward these electrodes. The negatively charged electrons move relatively rapidly toward a positively charged electrode (anode), but positively charged gas ions, being much heavier, migrate much more slowly toward the negatively charged electrode (cathode).

Rate of migration, indeed whether migration occurs at all, depends upon how much potential difference (voltage) exists between the electrodes. If the voltage is too low, the force of attraction to the electrodes is slight, and the ions move very slowly toward them. In such a setup there is time for positive and negative ions to recombine (neutralize); this process is known as recombination (Figure 11-2-1). In the case of two plates charged by a battery, the plates are in the process of collecting some of the ions produced, but others have recombined already.

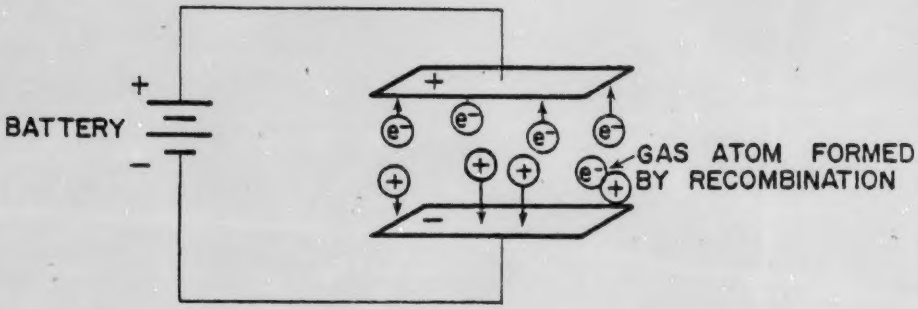

Figure 11-2-1. Migration and recombination of ions. Positively charged gas molecules (⊕) are attracted toward negative electrode; negatively charged electrons move toward positive charged plate. On the right, a neutral gas atom has been formed by recombination of an ion pair.

If voltage is initially very low, most ion pairs recombine, and a relatively small fraction of the total produced reaches the collecting electrodes. Thus, current flow at the electrodes is very small. As voltage is increased, a greater fraction of the ions is collected at the electrodes, until all are collected; at this point, saturation voltage has been reached. Further increases in voltage produce no additional ion collection - a current plateau is reached (Figure 11-2-2).

If voltage is further increased, at some point an increased current flow can be observed. At this point, additional ions are produced by the original ions, which are moving so rapidly toward the attracting electrode that they collide with other neutral gas molecules, causing them to become ionized (i.e., produce additional ions). This multiplication phenomenon is discussed in the later modules (Module 11-3, "Gas-Filled Detectors II: Proportional Counters," and Module 11-4, "Gas-Filled Detectors III: Geiger-

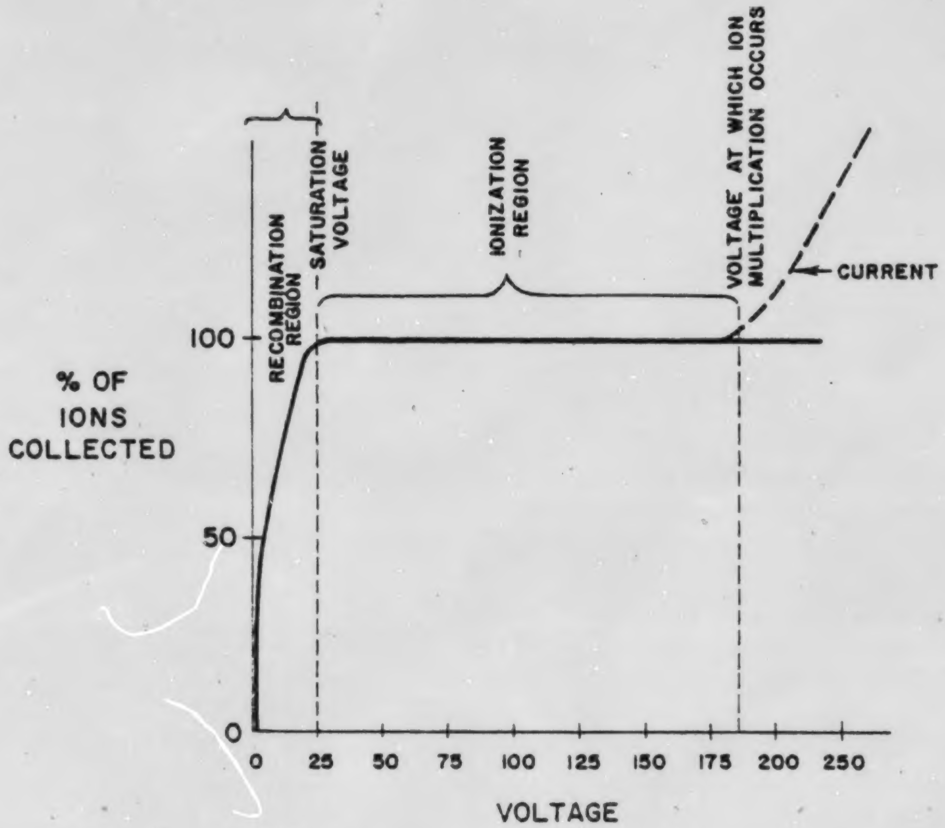

Figure 11-2-2. Typical saturation current curve: Ion collection efficiency is shown as a function of applied voltage between electrodes of an ionization chamber.

Mueller Counters"). Note: The voltage range below the saturation voltage is known as the "recombination region;" that between the saturation voltage and the voltage where the current begins to increase is known as the "ionization region." (Ionization chambers normally operate within this latter region.)

Basically, an ion chamber (Figure 11-2-3) consists of a gas-filled chamber containing two electrodes across which

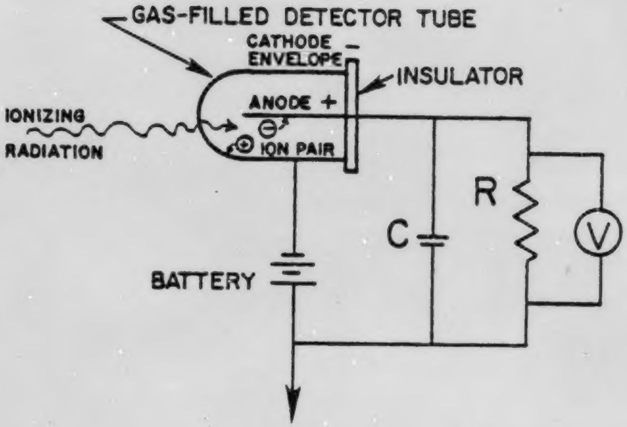

Figure 11-2-3. Simplified schematic of an ionization chamber circuit.

an electrical potential of a few tens to hundreds of volts may be impressed (exact voltage depends largely on size and shape of chamber, plus type of fill gas used, and its pressure).

Most commonly, a nonintegrating ionization chamber is cylindrically shaped, with a positively charged central electrode running the length of its central axis. The outer portion of the ionization chamber may be held at ground potential, making current flow along the central electrode convenient for measurement.

The production of ion pairs produces a flow of current (Figure 11-2-3) as electrons migrate to the central electrode, which is positive with respect to ground. Although this current could be measured directly, it is usually quite small; a large (high megohm) resistance inserted into the circuit allows voltage drop across it to be measured with a sensitive voltmeter. Note: Voltage and current are directly proportional, according to Ohm's law:

$$V = IR$$ Equation 11-2-1 (Equation = Eq.)

where

V = voltage, in volts.
I = current, in amperes.
R = resistance, in ohms.

By judicious selection of R, reasonable values of V can be obtained, permitting relatively easy and accurate measurements.

Also, capacitor C indicates (stray) capacitance (in farads). Response time of the instrument is a function of both the resistance and this capacitance, and is usually expressed as the time constant, which is simply the product of resistance and capacitance. If the value of RC is much greater than t (time required for collection of all the ions produced by a single ionizing event within the detector), the size of V can be calculated as follows:

$$V = \frac{Q}{C} \qquad \text{Eq. 11-2-2}$$

where Q is the charge collected (in coulombs), that is, the number of ions (produced by radiation) reaching the center electrode.

Equation 11-2-2 is used with integrating type ionization chambers; nonintegrating types have small values of RC time constant, and this equation is not valid.

The magnitude of ionization currents produced by nuclear radiations can be easily calculated. For example, ^{239}Pu emits alpha particles with energies of 5.15 MeV.

The number of ion pairs produced in a gas by these particles can be readily calculated by dividing energy of a particle by the average energy expended to produce one ion pair:

$$\text{I.P.} = \frac{E}{W} \qquad \text{Eq. 11-2-3}$$

where

I.P. = number of ion pairs.
E = energy of the particle, in eV.
$\underline{W}$ = average energy needed to produce one ion pair, or about 34 eV. (Note: W is underlined because it appears in Objectives as a term to be defined.)

EXAMPLE

Given: For a ^{239}Pu alpha particle of 5.15 MeV energy,

$$\text{I.P.} = \frac{5.15 \times 10^6 \text{ eV}}{34 \text{ eV/ion pair}}.$$

Find: Number of ion pairs.

Solution: I.P. = 1.5×10^5 ion pairs.

A single alpha particle produces about 150,000 ion pairs, or a total of 300,000 ions (cations and electrons). This total may seem like a large number of ions, but it is only enough to produce a small current. An ampere is defined as 1 coulomb per second. One coulomb is equivalent to 6.28 x 10^{18} electron charges. Thus, the coulomb production of a single ^{239}Pu alpha particle may be calculated:

$$\frac{1.5 \times 10^{5} \text{ ion pairs (electrons)}}{6.3 \times 10^{18} \text{ electron charges}} = 2.4 \times 10^{-14} \text{ coulombs.}$$

If one such alpha particle is produced each second, the current would be only 22 femtoamps (0.022 picoamps), which is immeasurable except with the most sensitive and delicate ammeters available today. Even a curie of ^{239}Pu produces only a current of about one milliamp.

As suggested above, a high-resistance element can help overcome this sensitivity problem. For example, an ion chamber instrument might be provided with a 100-megohm resistance. By using Ohm's law, voltage drop (pulse) across the resistor may be calculated:

$$\begin{aligned} V &= IR \\ V &= 2.4 \times 10^{-14} \text{ amperes} \times 1 \times 10^{8} \text{ ohms} \\ &= 2.4 \times 10^{-6} \text{ volts} \\ &= 2.4 \text{ microvolts/alpha particle.} \end{aligned}$$

Such voltages are relatively easy to measure, either in the form of individual pulses or as a constant potential difference.

By comparison with other radiations, alpha particles have a high specific ionization - that is, they produce many ions per unit of track length. Where an alpha particle may produce 150,000 ion pairs, a beta particle may leave only a few hundred while traveling the same distance. Thus, the beta particle deposits less energy and produces a much smaller pulse (Figure 11-2-4).

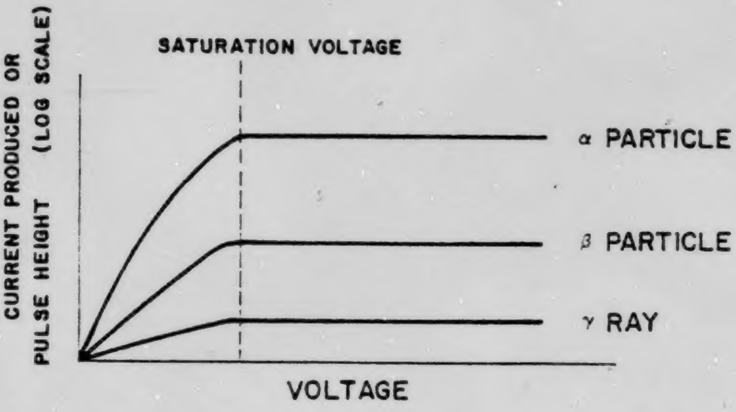

Figure 11-2-4. Schematic representation of current produced (voltage drop - pulse height - across a resistor) as a function of applied voltage in an ionization chamber.

As can be seen, gamma rays produce still smaller pulses. Note that saturation voltage does not change; it remains the same irrespective of the nature of the ionizing radiation.

One very important use for an ionization chamber is separation of different types of radiations by electronic means. If, for example, an instrument is needed that can respond only to alpha-particle-induced radiation, an electronic circuit can be designed to accept only those pulses exceeding minimum height. This circuit counts or accepts only those intense pulses associated with alpha particles, while discriminating against smaller beta- or gamma-induced pulses.

Such a circuit, called a discriminator, is useful in a counting laboratory for such applications as counting only alpha particles or, perhaps, one other kind of particle.

Sophisticated pulse-height discriminators, with upper and lower pulse size cutoffs, can also be used to allow only a limited range of pulse sizes to be passed into the counting circuitry; for example, pulses from beta particles only can be selectively counted.

The above application refers to the use of ion chambers as pulse or individual event counters. Another important application is measuring of exposure or exposure rate, or dose or dose rate.

Exposure is defined in units of charge. According to the internationally accepted definition, one roentgen equals 2.58×10^{-4} coulombs per kg of air; similarly, for 500 cm^3 (a typical chamber size) of air at standard temperature and pressure, one roentgen equals 1.7×10^{-7} coulombs. An exposure rate of 1R/hr equals 1.7×10^{-7} coul/hr, or 4.5×10^{-11} amp. By using Ohm's law (Equation 11-2-1), the voltage drop across a 100-megohm resistor may be calculated as follows:

$$
\begin{aligned}
V &= 4.5 \times 10^{-11} \text{ amperes} \times 10^{8} \text{ ohms} \\
&= 4.5 \times 10^{-3} \\
V &= 4.5 \text{ mV.}
\end{aligned}
$$

Voltages on the order of 4.5 millivolts can be measured readily with small, inexpensive portable meters. In fact, voltages 1000 times smaller can be measured with such meters. By providing several different resistors and a se-

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lector switch, a variable-range rate meter can be made.

Rate measurements are made with mean level, integrating type ionization chambers. The total exposure (dose) can be determined if the instrument is designed to measure total charge. To do so, a static voltage or charge is put across two electrodes in a gas (usually air) filled chamber. Under normal circumstances, gas does not conduct, and the potential difference remains stable for a long period of time; however, formation of ion pairs by radiation causes the voltage to be reduced, and this change in potential can be measured. Then, if capacitance of the system is known, the total amount of produced charge in the chamber can be calculated (Equation 11-2-2). This result can also be inferred from Ohm's law:

$$\Delta t \, \Delta V = \Delta Q \times R \qquad \text{Eq. 11-2-4}$$

where

ΔQ = change in charge in coulombs, or number of ion pairs produced in chamber.
Δt = fixed period of time, seconds.
R = resistance of the system, ohms.

Ordinarily, for integrating systems, capacitance C is known, so Equation 11-2-2 can be used.

IONIZATION CHAMBER DESIGN

Ionization chambers, being relatively simple devices, lend themselves to various geometrical configurations. They can be constructed to perform in a number of situations where other instruments might prove unsuitable. Also, their sensitivity can be increased by making the chamber larger. (Chambers as large as 40 liters have been routinely used for low-level radiation monitoring.)

Some ionization chambers are large and bulky and, because of their size, their electric field strength or voltage gradient at points far from the collecting electrode, may be too small to efficiently collect all electrons produced by the radiation. Ions produced far from the electrode also have a relatively long distance to travel and, therefore, have a greater chance to recombine before being collected.

Sensitivity can be altered easily by using a different fill gas or by changing the fill-gas pressure (Figure 11-2-5). Increasing gas pressure in a given volume means that more gas molecules are present, creating a greater likelihood of radiation interactions.

An important design consideration for any ionization chamber is its electrical insulation. The collecting electrode must be electrically insulated from the remainder of the circuit, else some of the current (charge) may leak off. This leak off may cause erroneously low readings. In the case of an integrating ion chamber, the system may partially or completely discharge, producing erroneously high results. Also, tiny quantities of dust, grease or other materials can produce undesired effects. Insulator performance may be degraded by exposure to chemical fumes or vapors, or to

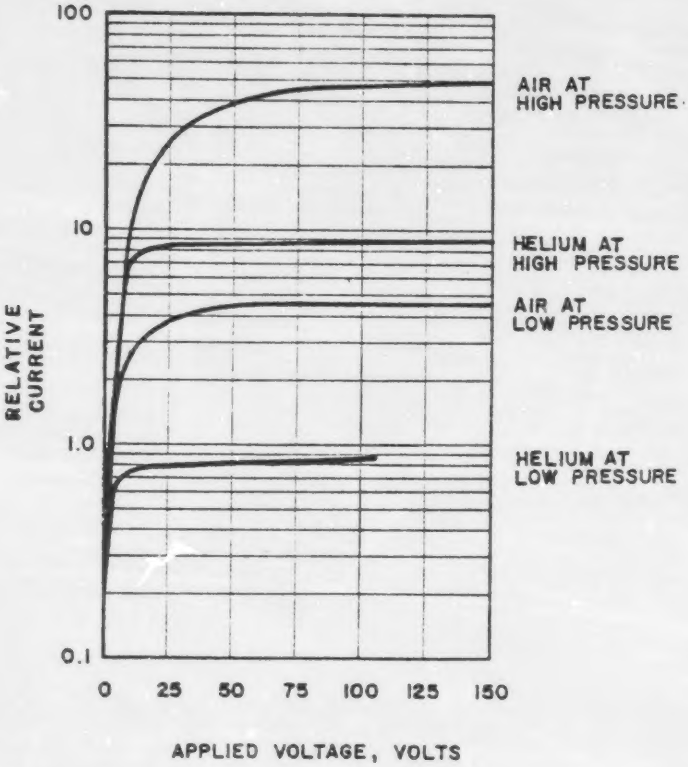


Figure 11-2-5. Current versus voltage relationships for various fill gases and pressures in a cylindrical ionization chamber. (Note: variation in saturation voltage is in the 0-25 V range.)

excesses of radiation. For example, polystyrene - normally a good insulator - degrades in radiation fields as low as a few R/hr.

Another important design consideration is chamber wall material and thickness; however, there is flexibility in this regard, depending on the radiation to be measured. For example, if exposure is to be measured, walls should be as nearly air equivalent as possible.

The term "air equivalent" indicates some material other than air, which responds as air does when exposed to radiation. Equivalence can be approximated by materials with average atomic numbers (Z) equal to the average atomic number of air, and with densities the same as air. Obviously, total equivalence is impossible. But various low-density plastics, such as Bakelite, are often used. (One commercial portable ion chamber dose rate meter even uses a styrofoam coffee cup for a chamber!)

The importance of wall materials cannot be stressed too greatly if an ion chamber is to be used to determine tissue dose or dose rate. If wall material differs too greatly in Z from that of tissue, a radically different response will be obtained when measuring doses or dose rates from x or gamma radiation with energies below a few hundred keV.

Overresponses of several hundred percent can be obtained easily, particularly for photons with energies near 50 keV. This energy dependence, which can be offset by proper selection of wall material, is attributable to increased numbers of secondary electrons produced by photon interactions in the chamber walls (when walls are made of materials with a Z higher than tissue). Many of these secondary electrons get into the ionization chamber proper and produce an overresponse.

Similarly, an underresponse can occur if the effective Z of the wall is lower than that of tissue.

Chamber wall thickness also affects ion chamber response. A thick wall can attenuate some of the radiation, particularly in the case of low-energy photons, or charged particles. Indeed, an ion chamber for detection of alpha particles must have a thin wall or a window through which the particles can pass. Similarly, if beta measurements are to be made, the chamber wall must be thin enough to allow passage without significant attenuation or energy absorption. For alpha particle measurements, the chamber wall must be very thin (minimum absorption). In many cases, chambers to detect alpha particles have thin mylar or plastic windows through which alpha particles may easily enter their chambers.

Ionization chambers are also subject to various environmental conditions that alter or affect their response. Dust, smoke or chemicals in the atmosphere may change response, or cause spurious readings of some chambers, particularly air ionization chambers that are not sealed. Atmospheric temperature and pressure changes may produce minor changes in response; however, sealed ionization chambers overcome many of these disadvantages to a great extent.

The design of ionization chambers is by no means simple, and the brief discussion above has only touched on some of the more significant and perhaps obvious aspects of design. Other factors to be considered include:

Physical shape of chamber.

Electrical field distribution inside the chamber.

Loss of collection efficiency due to edge or end effects (which may be caused by bowing or bending of electric field at edge of the chamber).

Also, recombination of ions (produced in a chamber from high-intensity bursts of radiation) may severely limit the capability of a chamber to measure pulsed fields (such as may be found around accelerators, field emission x-ray units, weapons bursts or pulsed reactors).

APPLICATIONS OF IONIZATION CHAMBERS

The applications of ionization chambers can be broadly lumped into these four categories:

Laboratory counting of radioactivity (pulse counters).
Radiation survey meters (mean-level detectors).
Dosimeters (integrating).
Special applications.

Those included in the first category are used as ionization pulse counters for determining individual ionizing events or, in some cases, for spectroscopy. Spectroscopy is a process by which the energy spectrum of a group of particles can be determined. Discrimination between alpha and beta particles, described above, is a crude and simple form of spectroscopy (used, in this case, to differentiate between two different kinds of radiations).

Spectroscopy is also useful for identifying specific radionuclides. In the spectroscopy process, pulses from an ionization chamber are electronically sorted and counted according to size; larger pulses are produced by the more energetic particles; smaller pulses are produced by the lower energy particles.

For any given particle (e.g., an alpha particle), pulse size or height is directly proportional to the particle's energy. That is, a spectrum may be obtained. What the spectrum (Figure 11-2-6) clearly shows is a mixture of two alpha

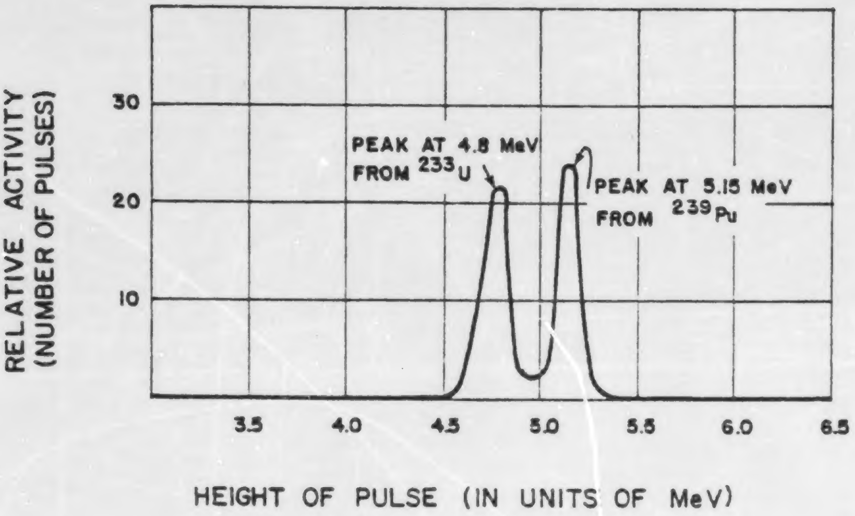


Figure 11-2-6. Alpha particle spectrum from an ionization counter spectroscopy system.

particles, one with an energy of 4.8 MeV and the other with an energy of about 5.15 MeV. Reference to proper decay schemes shows that ^{233}U emits an alpha particle with an energy of 4.8 MeV, and ^{239}Pu emits alpha particles with energies of 5.13 and 5.15 MeV. A mixture of ^{233}U and ^{239}Pu is inferred. If efficiency of the counter were known, it would be possible to compute the actual amount present of each radionuclide.

The term "efficiency," as applied to radioactivity counting instruments, refers to the fraction of the particles detected by the system.

Virtually all radioactivity counting applications of ionization chambers are for alpha particles. Occasionally, an ion chamber may be used for other kinds of radiations. But the relatively large pulses produced by alphas, plus the discrimination capability, makes ionization chambers particularly suited to alpha counting.

Chambers can be designed so that a tiny alpha source (or unknown preparation) can be inserted in a sensitive volume; thus, a counting efficiency of 100% can be achieved. This arrangement is known as "4π efficiency," or 4π geometry (Figure 11-2-7), because alpha particles from the source, emitted uniformly in all directions from the source (over the entire 4π steradians of a sphere) are captured.

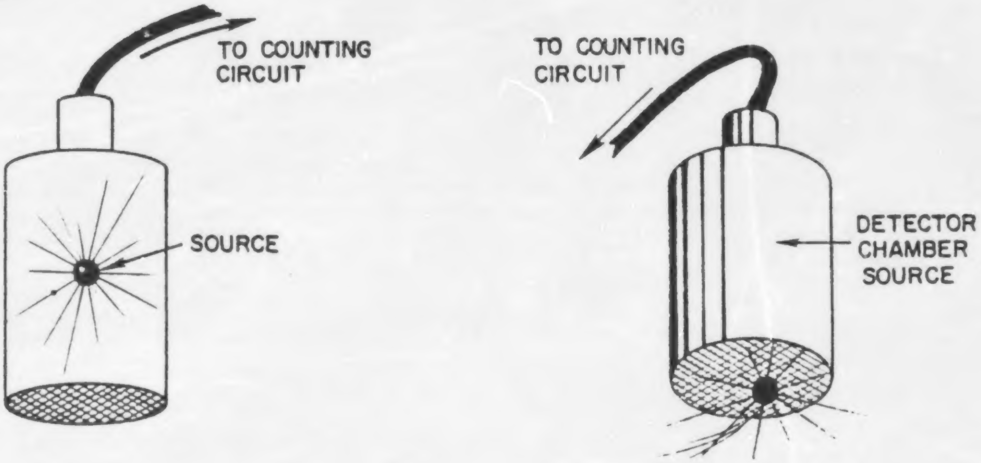


Figure 11-2-7. Geometrical considerations.

A second major ionization chamber use is as a portable survey instrument (for measuring radiation exposure rate or dose rate). The most common instrument is a pistol shaped, portable rate meter known as "Cutie Pie" (Figure 11-2-8).

The chamber is usually equipped with a removable shield over one end of its cylindrical, thin-window ionization chamber. With the shield off, the instrument responds to β

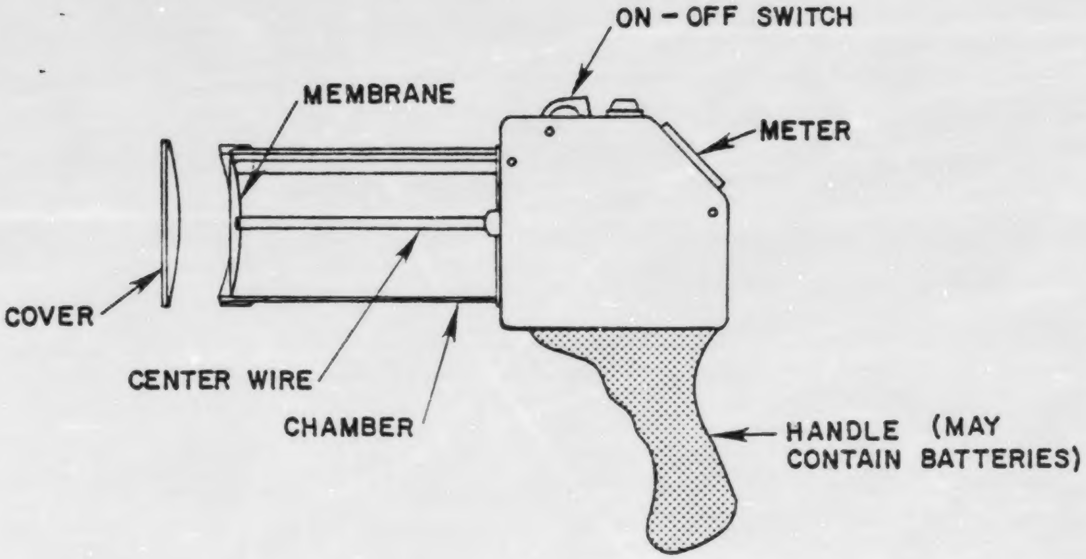


Figure 11-2-8. "Cutie Pie" rate meter.

radiation. With the shield in place, however, β radiation is wholly or partially screened out.

Typical chamber volume is about 500 to 600 cm^3; hence, an exposure rate of 1 mR/hr produces a current of about 2.5×10^{-14} amperes. This rate limits its lower usable range to about 0-3 mR/hr, even with the most advanced electronics.

Because most ionization chambers have plastic walls with an effective Z approximately equal to that of air, their energy dependence characteristics (Figure 11-2-9) are quite good; e.g., the Cutie Pie (Figure 11-2-8). For photons (x and gamma rays) with energies above 200 keV, the curve is essentially flat; that is, the response is independent of energy. The gamma energies of most fission products fall above 200 keV. But x-rays (as from laboratory apparatus or radionuclides) and scattered gamma rays usually

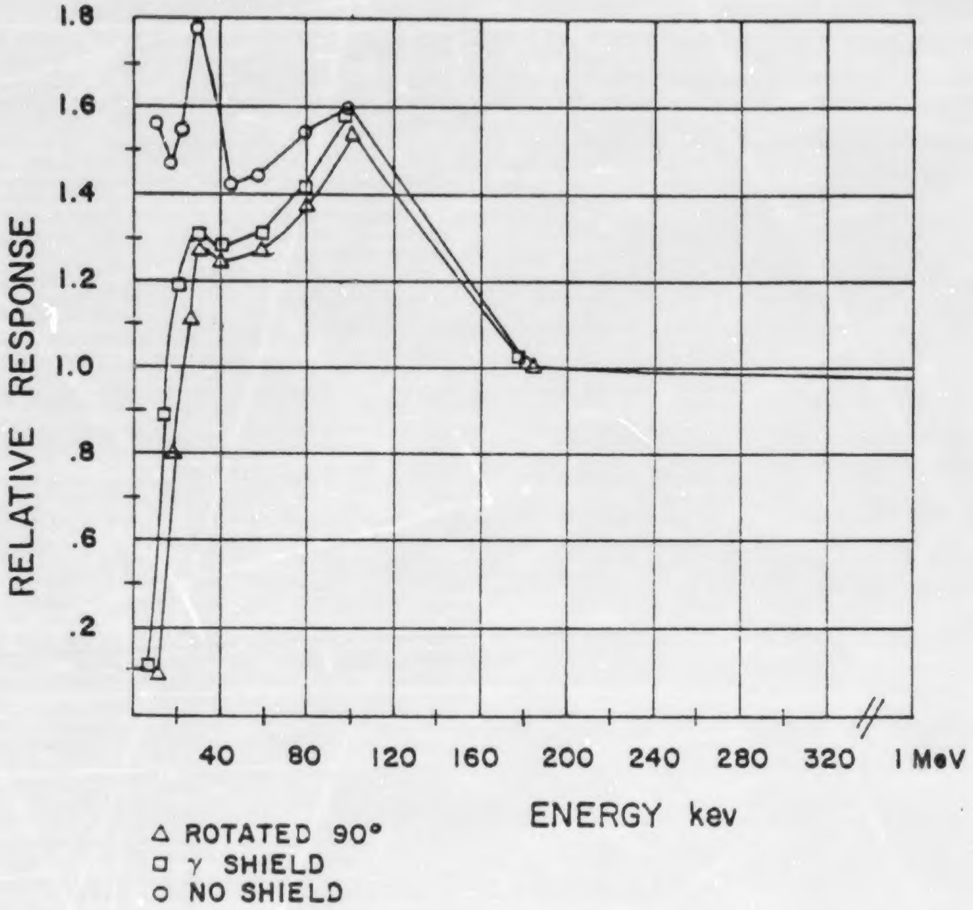


Figure 11-2-9. Energy dependence of the Cutie Pie shown in Figure 11-2-8.

have much lower energies, and their readings may be inaccurate.

There are many portable, battery-operated ionization type survey meters that do not have the typical gun shape of the Cutie Pie. Often these are called Cutie Pies - the name has become more or less generic for any ion chamber survey meter with fairly good energy response characteristics and an external, cylindrical chamber.

Other types of ionization chamber survey meters, including some with relatively small chambers containing gas under high pressure, permit higher degrees of sensitivity and lower exposure rates - as low as 0101 mR/hr - to be detected. These meters may have logarithmic readouts (rather than linear scales, as on the Cutie Pie), and are usable over an extremely wide range. Some commercially made units have a range from 0.01 mR/hr to 10,000 R/hr.

In recent years, Cutie Pie and other ionization chamber survey meters have been designed for use as dose rate meters or dose accumulation devices. When used as dose rate instruments, the ionization chambers are used as mean-level indicating devices, and the current (charge per unit time) produced within each chamber is measured or calibrated in units of dose rate - usually mrad/hr. The chamber may also be used as a charge accumulator; that is, the electronic circuitry can be designed to permit continuous totaling of charge (which is proportional to total dose received).

Dose meters, or dosimeters, make up the third major category of ionization chamber applications. In this application, the ionization chamber is used as a true charge accumulator, or integrating device, such as the gold-leaf electroscope.

A gold-leaf electroscope (Figure 11-2-10) is, in the broad and simple sense, an integrating ionization chamber. The gold-leaf electroscope essentially consists of two thin, gold foils attached to a conductor inside a sealed jar or chamber. When a charge is placed on the conductor, it is transferred to the gold leaves, which - because they now have the same electrical charge - repel each other. When the chamber is exposed to ionizing radiation, the ions produced neutralize the charge on the gold leaves, allowing

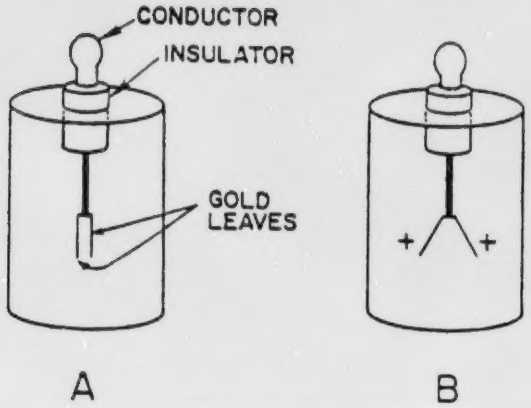


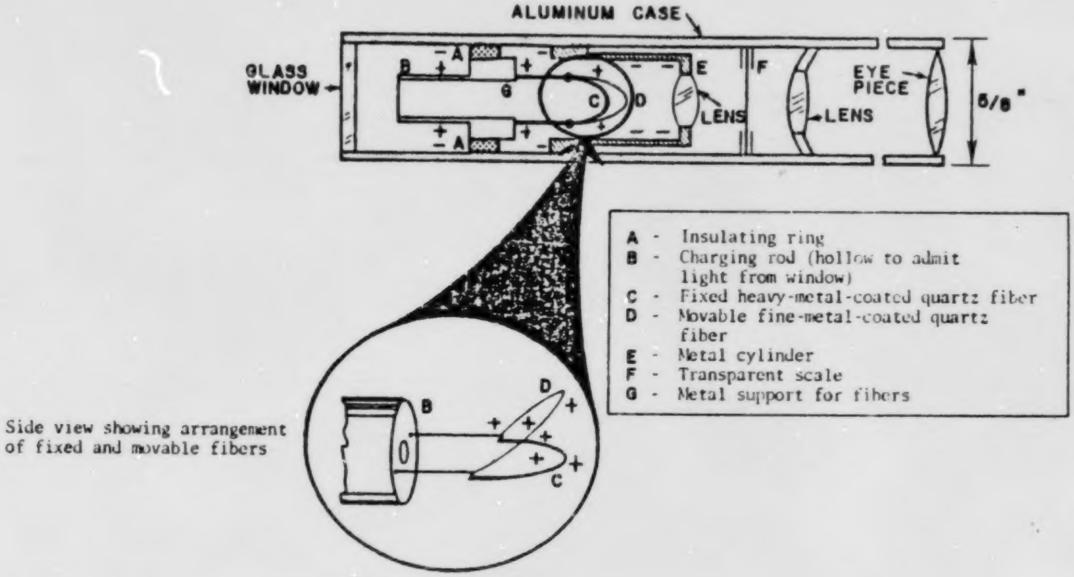
Figure 11-2-10.
Schematic drawing of a simple gold-leaf electroscope.
Drawing A shows the device, with essential parts labeled, in an uncharged condition.
Drawing B shows the device in a charged condition.

gravity to draw them back down into the vertical position.

The principle of the gold-leaf electroscope forms the basis for the pocket dosimeter or pocket ionization chamber (PIC). The dosimeter has a fine, metal-coated, quartz fiber about 0.25 inch (6 mm) long. This fiber is attached to a rigid metal electrode and suspended within a tiny gas-filled chamber. When a charge is put on the electrode, the fiber is given the same charge and, therefore, draws away from the electrode (electrical repulsion). The amount of voltage placed on an electrode determines how much repulsion occurs; i.e., how far away the fiber moves.

When the dosimeter is exposed to radiation, ions produced in the chamber reduce the voltage (see Equation 11-2-4) or potential between the fiber and the electrode, and the flexible fiber moves back toward the rigid central electrode. How far it moves back depends on how many ions (how much radiation) is exposed to the instrument.

The amount of discharge (voltage change) can be measured with a voltmeter. Or a self-reading instrument can be equipped with a built in microscope-and-scale calibrated in appropriate exposure or dosage units (Figure 11-2-11). Those without microscopes are known as nonself-reading dosimeters.



Side view showing arrangement of fixed and movable fibers

A - Insulating ring
B - Charging rod (hollow to admit light from window)
C - Fixed heavy-metal-coated quartz fiber
D - Movable fine-metal-coated quartz fiber
E - Metal cylinder
F - Transparent scale
G - Metal support for fibers

Figure 11-2-11. Self-reading pocket ionization chamber (PIC).

If capacitance of the dosimeter is known, along with its charging voltage, Equation 11-2-2 can be used to calculate exposure to the chamber.

EXAMPLE

Given: A 1 cm^3 chamber with a capacitance of 100 pfd, charged with 130 volts. After exposure, voltage = 110 volts. The amount of charge produced in the chamber produces a change in potential of 20 V or,

$$V = \frac{Q}{C}$$

$$20 \text{ V} = \frac{Q}{1 \times 10^{-10} \text{ farads}}$$

$$Q = 2 \times 10^{-9} \text{ coulombs.}$$

Since the chamber volume is 1 cm^3, air density at standard temperature and pressure is equal to 1 cm^3 x 0.001293 gm/cm^3, or 1.3×10^{-3} g of air, or

$$\frac{2 \times 10^{-9} \text{ coul}}{1.3 \times 10^{-3} \text{ g}} = 1.5 \times 10^{-6} \text{ coul/g}$$

or 1.5×10^{-3} coul/kg of air.

The roentgen, as mentioned previously, is defined as 2.58×10^{-4} coul/kg.

Find: Exposure to chamber.

Solution:

$$\frac{1.5 \times 10^{-3} \text{ coulombs/kg of air}}{2.58 \times 10^{-4} \text{ coulombs/kg of air/R}} = 5.8 \text{ R.}$$

The final category of application for ion chambers (i.e., special applications) covers a variety of uses, but most importantly includes radiological or exposure standardization measurements. Standardization measurements - highly accurate and precise measurements - are accomplished with very precisely made instruments.

One such instrument is an integrating device very similar to a nonself-reading pocket ionization chamber. In this special situation, however, the ionization chamber is very carefully constructed, and its uniform walls are of appropriate thickness to permit energy measurements. It incorporates other features as well, including a special readout apparatus - essentially, a highly accurate electrostatic voltmeter.

Such chambers are sometimes known as condenser chambers or condenser R-meters, or just plain roentgen meters or R-meters. (Condenser is an old-fashioned word for capacitor, and the basic collecting device is merely an ion chamber connected across a capacitor.)

Another standardization instrument is the free-air or air-wall ionization chamber (Figure 11-2-12). In the free-air chamber, all the ions produced in a very precisely defined volume of air are collected. The device is constructed so that its "walls" are air; hence there are no errors from secondary electrons produced by other interactions in the surrounding air. Guard rings perfect the electric field within the chamber. And an electrometer, or very sensitive ammeter, directly determines the current.

A final, special application is the use of ion chambers to measure concentrations of radioactive gases. A small pump pulls air (or other gas mixtures) through the ionization chamber, and current produced in the chamber is mea-

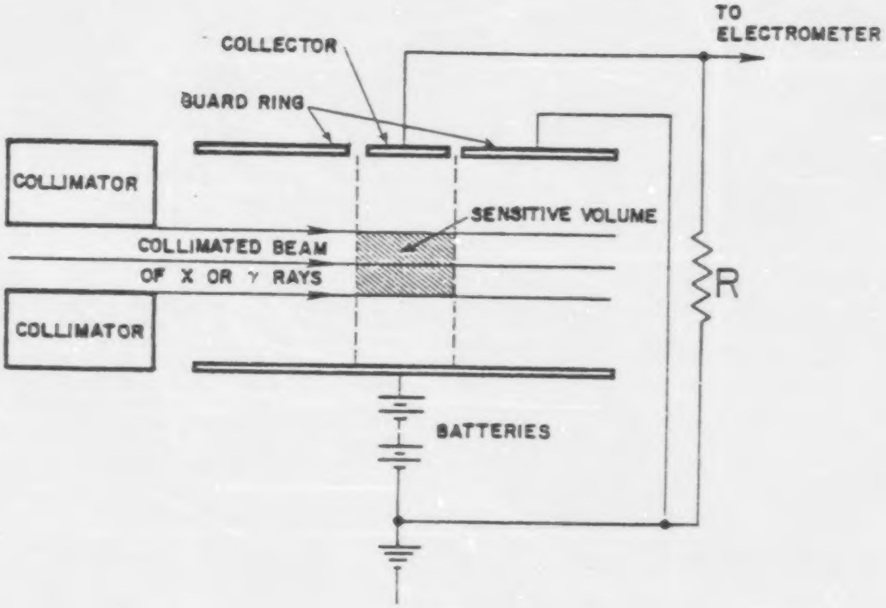

TO ELECTROMETER
COLLECTOR
GUARD RING
COLLIMATOR
SENSITIVE VOLUME
COLLIMATED BEAM
OF X OR γ RAYS
R
COLLIMATOR
BATTERIES

Figure 11-2-12. Schematic representation of a free-air ionization chamber.

sured directly with an electrometer. Such instruments are sometimes known as "flow-through ionization chambers." A common use is the measurement of tritium in air, and such devices are frequently called "T-sniffers" or "tritium sniffers."

EXERCISES

1. Determine how voluminous an ionization chamber filled with air would have to be to give a full-scale measurement of 10 R if capacitance of the chamber is 0.004 microfarads and voltage drop is 80 volts.

2. Refer to Figure 11-2-6; give approximate ratio of ^{239}Pu to ^{233}U.
3. Refer to Figure 11-2-9; determine true exposure rate for an indicated rate of 1.4 R/hr for ^{241}Am, which emits 60 keV photons.
4. Could a meter with a range of 0-10 volts, with a volume of 4 liters and a resistance of 1.3 megohms, be used for an ionization chamber survey meter if it were desired to measure an exposure rate of 40 mR/hr?

MATERIALS

1 ionization chamber (Cutie Pie).
1 variable voltage supply (or supply of batteries).
1 small source of radiation.
1 electrometer.
Linear graph paper.

PROCEDURES

1. Connect voltage supply (or battery) to ionization chamber.
2. Start with a very low voltage; measure current output

of chamber when exposed to source at a fixed distance; record results on Data Table 11-2-1.

3. Without changing source-detector geometry, increase voltage stepwise and again determine current. Repeat this procedure several times until current output becomes constant. This constant is the "plateau."
4. Continue to increase voltage until current begins to increase again. At this point, chamber is no longer functioning as an ionization chamber. No more measurements are required.
5. Plot the readings (taken in 2 and 3, above) as a function of voltage; from this plot determine voltage at which plateau begins. This point indicates "saturation voltage." Current in the plateau region is also the saturation current, and can be used to calculate average exposure rate to the chamber (if the volume is known). Determine the following:
    - Voltage required for saturation, in volts.
    - Saturation current, in amps.
    - Plateau length, in volts.
    - Calculated average exposure rate to chamber.
    - True or theoretical exposure rate.
    - Difference between calculated and true exposure rate; list some possible reasons for this difference.
6. Use laboratory system operating in ionization range to find pulse heights for α, β, γ. (Verify Figure 11-2-4.)
7. Verify Figure 11-2-5, if possible.
8. Demonstrate 2π, 4π geometry.
9. Demonstrate electroscope.

DATA TABLE 11-2-1

Date ____________________

Source type and activity ________________________________

Ionization chamber model ________________________________

Voltage	Current

REFERENCE MATERIALS

Glasstone, Samuel, ed. Sourcebook on Atomic Energy. 3rd ed. New York: Van Nostrand-Reinhold, 1967.

Howell, W.P. and Kathren, R.L. "Calibration and Field Use of Ionization Chamber Survey Instruments." In Health Physics Operational Monitoring, Vol. 2. Edited by C.A. Willis and J.F. Handloser. New York: Gordon and Breach, 1972.

Maushart, R. Radiation Protection Instrumentation. Oxford: Pergamon Press, 1972.

Price, W.J. Nuclear Radiation Detection. 2nd ed. New York: McGraw-Hill, 1964.

TEST

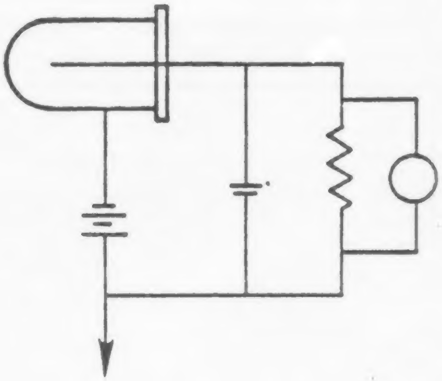
Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. Differentiate between mean level detectors and pulse counters.

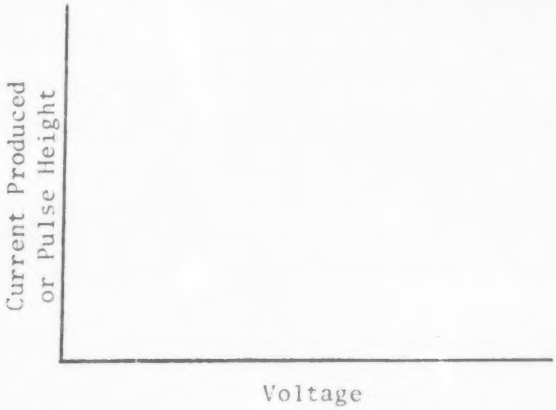
2. Draw a saturation curve for an ionization chamber. Label the recombination region, ionization region, saturation voltage and voltage at which multiplication occurs.

3. On the sketch of ionization chamber circuit below, label the following:

    1. the entry of incoming radiation
    2. gas filled detector tube
    3. insulator
    4. capacitor, resistor, voltmeter
    5. cathode envelope
    6. anode



4. Fill in the graph below to show that the ionization chamber can separate Gamma rays, Beta particles and Alpha particles. Label saturation voltage for each type of radiation. (Be sure your graph correctly shows what happens to saturation voltage as the nature of the ionizing radiation changes.)



Schematic Representation of Current Produced as a Function of Applied Voltage in an Ionization Chamber

5. What is the purpose of a discriminator?

6. What are ion pairs?

7. Briefly describe the migration of and collection of ion pairs when using an ionization detector.

8. What is the cause of recombination?

9. What is saturation voltage?

10. What does the term W denote in the equation I.P. (number of ion pairs) = $\frac{E}{W}$ (energy of the particle in eV.)?

11. What does the term "air equivalent" indicate? When is air equivalence important? How is equivalence approximated?

12. What is energy dependence? What are the consequences (effects) of energy dependence? How can energy dependence be prevented?

13. What is spectroscopy? For what major purposes is spectroscopy used?

14. Why are ionization chambers particularly suited to Alpha counting?

15. When an ionization chamber is used as a dose meter, what function does the ionization chamber serve?

16. When the ionization chamber is used as a portable survey instrument, exactly what does it measure? ____________________

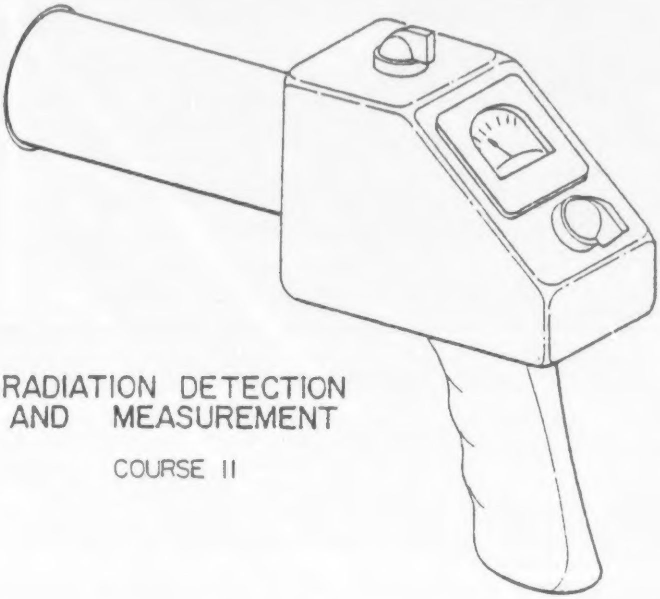
17. List two special applications of an ionization detector.

    a. ____________________

    b. ____________________

NUCLEAR TECHNOLOGY

MODULE 11-3 GAS-FILLED DETECTORS II: PROPORTIONAL COUNTERS.



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
⇨ MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

In the previous module, a category of gas-filled radiation detectors, called "ionization detectors," was described. A second major category of gas-filled detectors includes proportional counters.

In the previous module, it was shown that, once saturation has been reached, current output of an ionization chamber remains constant, even if voltage is increased over a fairly wide range. But if voltage is increased further, current output for a given dose or exposure rate begins to increase, and the chamber no longer operates as an ionization chamber: it becomes a proportional counter.

If the height or size of an output pulse from a single ionizing event in a chamber is proportional to particle energy, the instrument is a proportional counter. Relatively small and feeble pulses, such as are obtained with an ionization chamber, are greatly enlarged (by "gas amplification"), allowing differences in pulse size to be more easily differentiated. Also, simpler and less expensive amplifiers and other electronic apparatus may be used.

As with ionization chambers, proportional counters can be used as either individual event (pulse) counters or mean-level current-detection systems. But, proportional counters cannot be used as integrating or total dose measurement systems.

In this module, both the theory and applications of proportional counters are discussed, and the many similarities and differences between proportional counters and ionization chambers should become apparent.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Prepare a simple block diagram of a proportional-counter counting circuit.
2. Describe "gas amplification" and explain its role in proportional counting.
3. Cite several advantages of proportional counters over ionization chambers.
4. Differentiate between internal, external, gas-flow, pressurized and 4π proportional counters used for pulse counting.
5. Calculate the percent of counts lost in any proportional counter, given the resolving time and count rate.
6. List the five basic types of neutron proportional counters and briefly describe each type.
7. Explain why a BF_3 proportional counter is relatively good for making neutron measurements in a mixed neutron-and-gamma radiation field.
8. Define the following terms:
   P-10 gas.
   Internal proportional counter.

4π or absolute counter.
Flow-through counter.
Discriminator.
Resolving time.
Fission counter.
Proton recoil counter.
Hurst counter.
Hurst dosimeter.

9. Given the appropriate equipment or data, obtain the plateau of a proportional counter and use it to differentiate between alpha and beta particles in the laboratory.

GENERAL PRINCIPLES

If potential (voltage) across an ionization chamber is raised beyond a certain critical point, the output pulse from any given ionizing event will be increased in size (Figure 11-3-1). At this point, the chamber is no longer an ionization chamber: it is amplifying the current pulse. Further increases in potential increase this amplification.

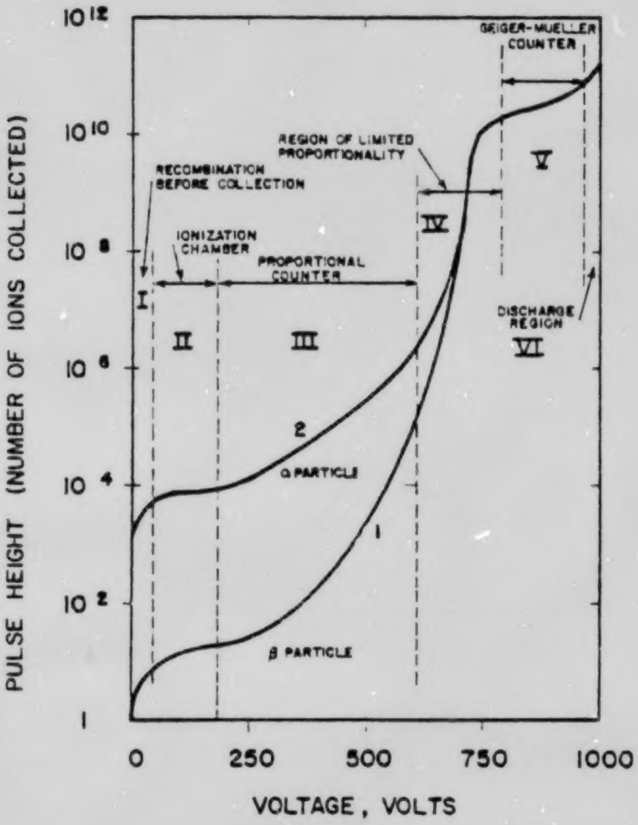

Figure 11-3-1. Pulse height versus applied voltage for a typical gas-filled detector, illustrating the various regions of operation. The proportional counter operates in Region III. Curve 1 is for a typical β particle and Curve 2 for an α particle.

The amplification occurs when the electric field strength, or potential gradient near the positively charged electrode, becomes so large that any electrons (produced by an ionizing particle) move toward it at a very high speed. In the proportional region (Region III) the speed of these electrons is so great that they bump into atoms and molecules of gas, causing some of these atoms and molecules to ionize also. Thus, an electron from each original ion pair may produce a large number of secondary ion pairs. This multiplication, or "avalanche," of electrons moving rapidly toward the positively charged electrode is known as "gas amplification" (Figure 11-3-2).

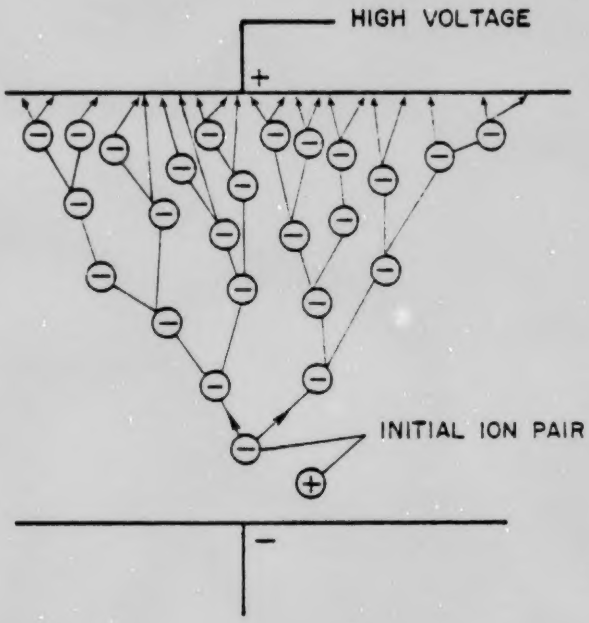

Figure 11-3-2. Schematic representation of gas amplification.

(Gas amplification was first investigated and described by J.S. Townsend, a physicist working in J.J. Thomson's laboratory at Cambridge about 1900. In his honor, gas amplification is sometimes known as the Townsend avalanche or Townsend cascade.)

In ionization chambers, there is no avalanche or multiplication effect, and hence the multiplication (gas-amplification) factor equals 1. In proportional counters, the gas-amplification factor can be made very large - as great as 10^5 or 10^6 - by increasing detector voltage. This larger voltage not only attracts electrons from farther away, but also causes them to accelerate faster. A normal gas-amplification factor is about 10^4.

Gas amplification can also be increased or decreased by detector design. A cylindrical tube, with a thin central wire serving as its positive electrode, best increases gas amplification. The center electrode (anode) must be very thin because only a thin wire can produce the relatively large electric field needed in the immediate vicinity of the wire.

Modern proportional counters use wires with diameters of 0.001" (0.025 mm) - called "1 mil wire." In older proportional counters, wire diameters of 0.02" (0.5 mm) to even 0.05" (0.13 mm) were common. Such thicknesses were needed to achieve the required strength with the alloys then in use.

Fill gas also influences gas multiplication. Air is seldom used - a relatively high voltage is needed, and oxygen tends to scavenge electrons, thereby reducing gas amplification and increasing collection time.

Water vapor hampers performance of a proportional countter - it, too, acts as an electron scavenger. In addition, water molecules tend to adhere to metal surfaces in the chamber, adversely affecting conductivity.

Simple gases, such as hydrogen, argon and nitrogen, either singly or in mixtures, may be used as fill gases; however, each of these suffers some major drawback: any discharge produced in them is apt to be perpetuated by ultraviolet light (emitted from cathode when positive ions are neutralized). Ultraviolet light may cause additional ionization, which can continue the discharge. This effect can be eliminated, however, by using lower voltages or by adding small amounts of polyatomic gases, such as methane.

A most popular proportional counter gas is a mixture of 10% methane and 90% argon, called P-10 gas. Methane favors stability of operation, while argon favors high gas amplification. Another popular mixture is 4% isobutane and 96% helium.

Proportional-counter response is also affected by fill-gas pressure. Increased pressure provides more gas molecules per unit volume, with increased sensitivity, but higher voltage is required to achieve a similar level of gas amplification. For example, a counter, at atmospheric pressure, may provide a gas-amplification factor of 100 when operated at 750 volts; after filling the same counter with gas at a pressure of four atmospheres, the number of available gas molecules is increased four; but, to get the same gas-amplification factor, 1500 volts are required.

The electronic circuitry of a proportional counter is very much like that of an ion chamber. Because of the larger output pulse (or current, in the case of a mean-level detector), it is not necessary to use a preamplifier. Also, highly sensitive ammeters (electrometers) are not required.

Proportional-counter readout can be accomplished with relatively simple and standard electronic components. Therein lies one of its great advantages and economies. For a pulse counter, associated electronics need only have a sufficiently rapid response to resolve or count each pulse from the proportional counter. Pulse duration, though longer than that of an ionization counter, still is very short, usually in the neighborhood of 0.2 to 0.5 μsec.

There is an upper limit on the counting rate of any proportional pulse counter; for example, by assuming a pulse duration of 0.2 μsec and instantaneous recovery time, the system detects:

$$\frac{1}{2 \times 10^{-7}\ \text{sec}} = 5 \times 10^{6} \text{ pulses (ionizing events) per second.}$$

Actually, this upper limit is a function of how long it takes to collect the slower moving positive ions. This period may be as long as 100 μsec; usually, however, it is on the order of a few microseconds.

In theory a proportional detector can be used as a mean-level current measurement device. But a very stable high voltage is required because even a slight change in voltage produces a great change in gas-amplification factor (Figure 11-3-1). For this reason, proportional counters are almost never used as mean-level devices (even with the highly stable power supplies now available). For all practical purposes, the proportional region is used only for pulse counting, hence the name, "proportional counter."

APPLICATIONS TO ALPHA AND BETA COUNTING

The relative simplicity and inexpensive nature of the proportional counter and associated electronics make this system widely acceptable for laboratory uses. Designs of these devices range from simple sealed cylinders to windowless or open proportional counters in which gas flows constantly through the detector. Proportional counters usually fall within one or more of the following classifications:

Internal.
External.
Gas-flow.
Pressurized.
Windowless.
Absolute (4π).

A sample is inserted directly inside the chamber of an internal proportional counter (Figure 11-3-3) which greatly

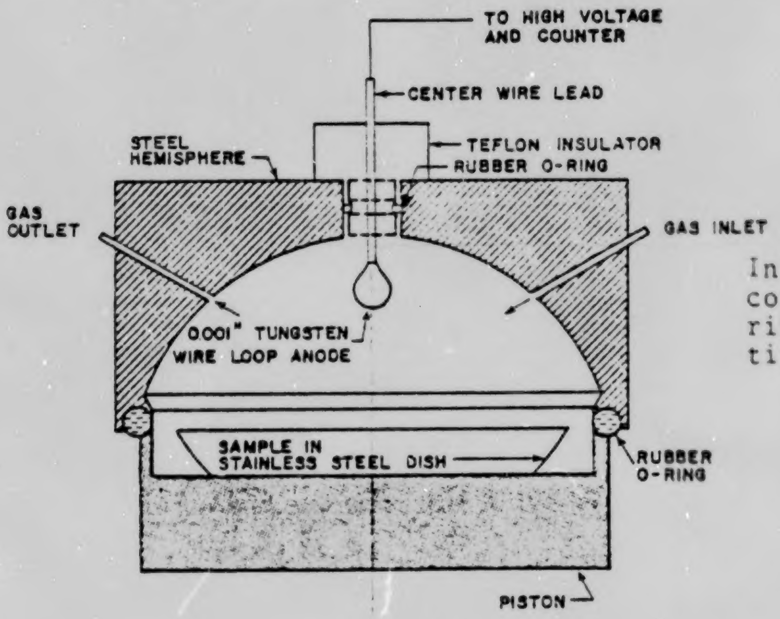
TO HIGH VOLTAGE AND COUNTER
CENTER WIRE LEAD
STEEL HEMISPHERE
TEFLON INSULATOR
RUBBER O-RING
GAS OUTLET
GAS INLET
0.001" TUNGSTEN WIRE LOOP ANODE
SAMPLE IN STAINLESS STEEL DISH
RUBBER O-RING
PISTON

Figure 11-3-3. Internal proportional counter chamber. O-rings provide gas-tight seal.

increases counting efficiency. If the sample is suspended in the middle of the counting chamber, enabling all particles to be collected, the device is called an absolute or 4π counter (Figure 11-3-4).

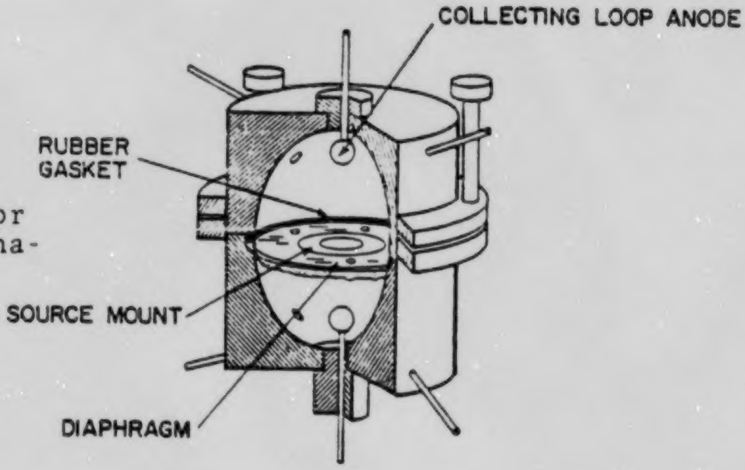

Figure 11-3-4.
4π geometry counter for absolute beta determination.

In an external counter, the sample is outside the gas-filled pressurized chamber.

Windowless counters (Figure 11-3-5) may be open to the atmosphere, but are always gas-flow or flow-through, internal-type in design. In the flow-through counter, gas continually flows through the chamber, constantly flushing it. The flow is extremely low, however, so that gas consumption is nominal. A positive pressure seal, in the case of a windowless counter, protects against pinhole leakage through a thin window.

After each use, the chamber must be purged with a large

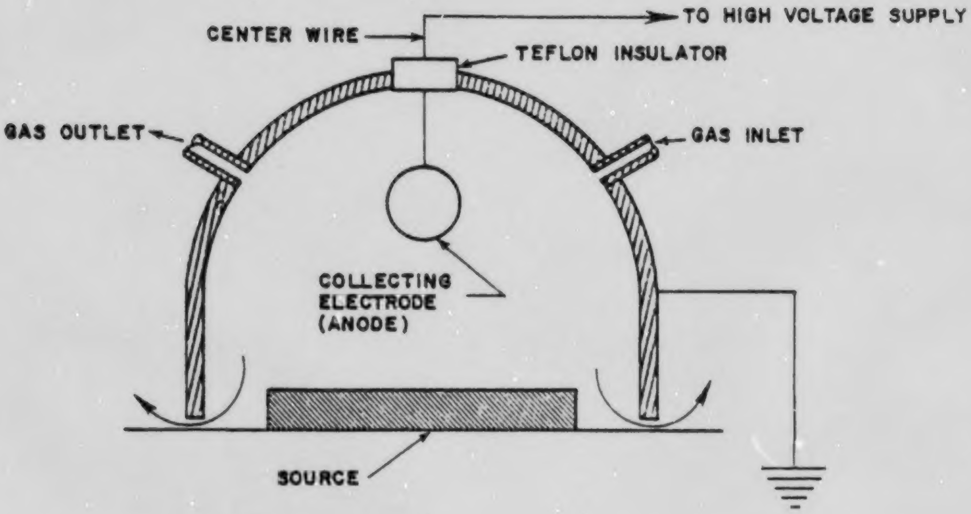
TO HIGH VOLTAGE SUPPLY
CENTER WIRE
TEFLON INSULATOR
GAS OUTLET
GAS INLET
COLLECTING ELECTRODE (ANODE)
SOURCE

Figure 11-3-5. Windowless counter, open to atmosphere. Arrows indicate gas leakage paths. Windowless counters, open to atmosphere, are rare and wasteful of counting gas.

flow of gas for a brief interval to remove any residual air or other contaminants.

Flow-through windowless counters are widely used for alpha and beta counting. For alpha particles, a large pulse is obtained - electronic discrimination can be used to eliminate smaller pulses.

The internal proportional counter has a characteristic counting curve with two plateaus (Figure 11-3-6). Note that the counting rate (not the current) is plotted. Also, starting voltage for the alpha plateau is that voltage at which the most energetic alpha particles give up all their energy in the chamber. Gas multiplication produces a pulse of sufficient size to pass the discriminator. (A discriminator, described in Module 11-2, is a circuit that permits only pulses

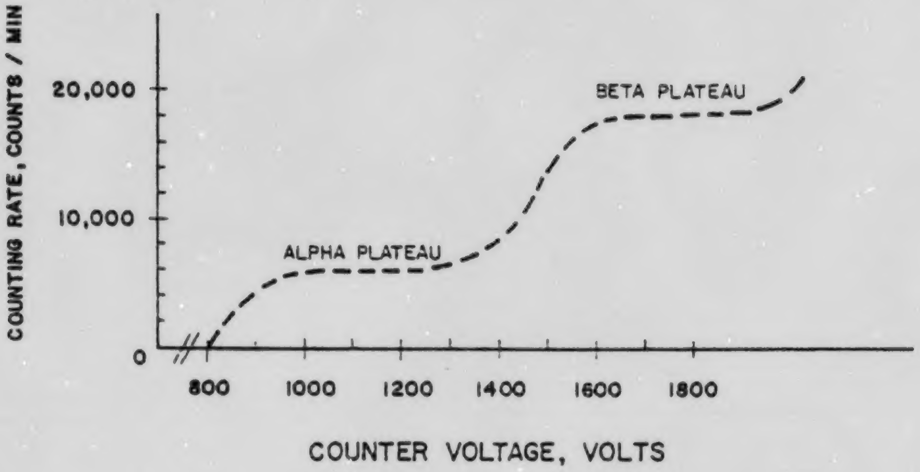

Figure 11-3-6. Plateaus for typical proportional counter.

greater than a certain size or voltage to pass through into the next part of the circuit.) Further increasing voltage increases pulse size but not the counting rate. At some point, however, the counting rate rises again because of the amplification of lower-energy particles or ionizing events (as from beta particles). In this way the alpha plateau is determined. Note: Proper operating voltage is in the center of this plateau.

Extending voltage even higher produces a beta plateau (which is not really flat, but has a slope, because not all beta particles have the same energy). Typically, because of the discrimination feature, background counting rates will be low - less than 1 count per minute for alphas and perhaps only up to 50 counts per minute for betas. The alpha background may come from alpha emitters present in trace quantities in the walls of the chamber or from cosmic rays in rare instances. The beta background includes the alpha background, but is mostly from external cosmic and gamma-ray sources.

In the internal proportional counter (Figure 11-3-7), the preamplifier may or may not be needed. Also, the amplifier must be highly linear (i.e., it must have the same amplification or gain for a given setting even if pulse size varies).

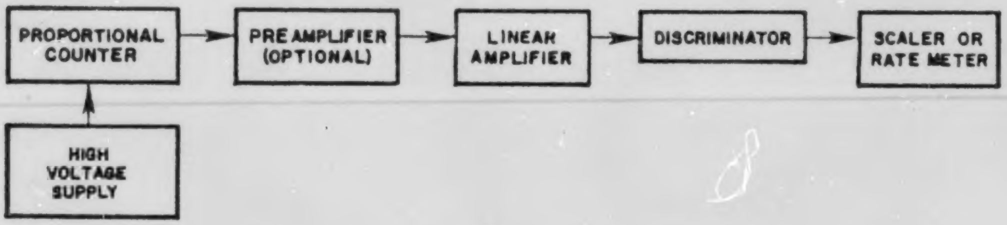

Figure 11-3-7. Block diagram of a proportional counter circuit.

The internal proportional counter enjoys one great advantage over other designs: no absorption of particles occurs in the window. In addition, placing the sample in the counting chamber reduces errors caused by geometry and backscattering. For these reasons, internal proportional counters are often used for absolute or precision counting.

Also, a relatively small resolving time permits very high counting rates with no loss of accuracy. Resolving time is the time it takes for the instrument to count a pulse and then recover to count the next one; in other words, it is the minimum time lapse between interactions of two particles within the detector. Resolving time, if any, can be a severe restricting factor in useful counting. Typically, proportional counters have resolving times on the order of msec or less, which enables them to count up hundreds of thousands of events per minute with no appreciable loss.

Resolution losses can be calculated as follows:

$$\% \text{ loss of counts} = 100 \times \text{counting rate} \times \text{resolving time}$$

Equation 11-3-1
(Equation = Eq.)

and, for a counter with a resolving time of one microsecond, a loss of 1% would occur at a counting rate of 600,000 counts per minute (cpm).

Proportional counters are also used in the field as portable survey meters (Figure 11-3-8) to detect alpha contamination on surfaces. Such proportional counters may be of the flow-through or air-proportional types, with detectors

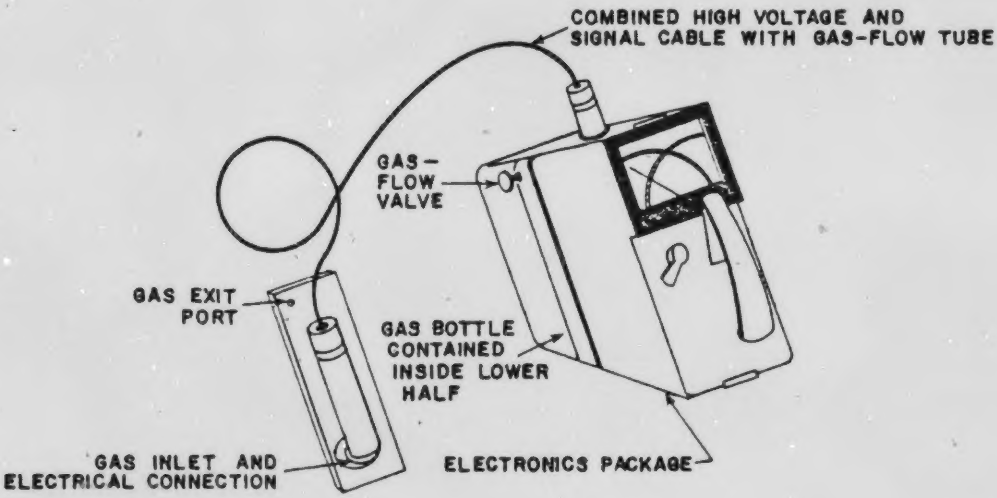
COMBINED HIGH VOLTAGE AND SIGNAL CABLE WITH GAS-FLOW TUBE
GAS-FLOW VALVE
GAS EXIT PORT
GAS BOTTLE CONTAINED INSIDE LOWER HALF
GAS INLET AND ELECTRICAL CONNECTION
ELECTRONICS PACKAGE

Figure 11-3-8. Commercial gas-flow proportional counter adapted as a portable alpha survey meter.

(chambers) made in the shape of a flat plate. This shape covers a relatively large surface area, perhaps as much as 100 cm^2. Portable gas-flow proportional counters contain their own gas supplies in small interchangeable metal bottles filled with gas under pressure.

NEUTRON PROPORTIONAL COUNTERS

The proportional counter is extensively used for detecting and measuring neutrons. Remember, neutrons do not ionize directly but cause ionization indirectly within these five basic types of neutron proportional counters:

BF_3 counters.
Boron-lined counters.
^3He counters.
Fission counters.
Proton recoil counters.

BF_3 COUNTER

The BF_3 counter is most common. It finds application in both portable, battery-operated instruments and fixed, ac-powered units. In a BF_3 counter, the detector is filled with BF_3 gas, which may be enriched in ^{10}B, a strong neutron absorber that produces alpha particles as shown below:

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{7}_{3}Li + ^{4}_{2}He + 2.78\ MeV.$$

Thus, a ${}^{10}B(n,\alpha){}^{7}Li$ reaction produces an alpha particle (${}^{4}_{2}He$) inside a BF_3 proportional counter, along with an MeV ${}^{7}Li$ nucleus. These two particles, because they are produced simultaneously, produce a single pulse.

BF_3 counters have been constructed in virtually all sizes and shapes. Most common is the cylinder, which - typically - is about 6 inches (150 mm) long and about 0.75 inch (19 mm) in diameter. Ordinarily, 96% enriched ${}^{10}BF_3$ gas is used under a pressure of several atmospheres. Because of the gas used and the pressure, operating voltage of the tube may be relatively high - in the neighborhood of 1500 to 2000 volts.

While BF_3 counters are highly useful, their sensitivity varies as a function of neutron energy (Figure 11-3-9).

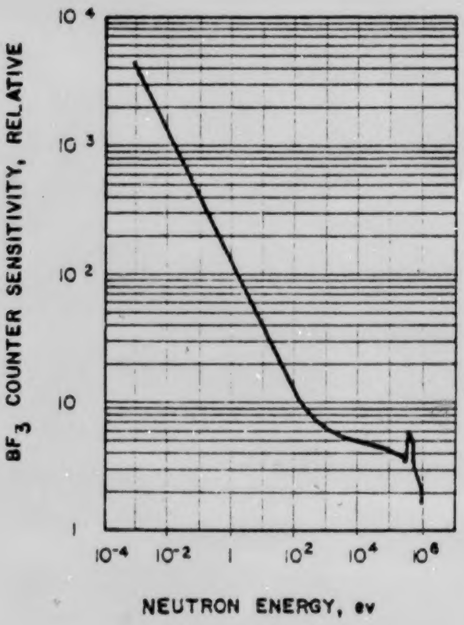

Figure 11-3-9.
BF_3 counter sensitivity or efficiency.

To increase response of the BF_3 tube to intermediate- or high-energy neutrons, it is sometimes wrapped in polyethylene, paraffin or other hydrogenous material to slow down the neutrons (i.e., decrease their energies). A thickness of about 4 inches of paraffin (100 mm) is needed to make responses approximately (within a factor of 2) independent of neutron energies over a wide energy range. A heavy, complex shield is required to make a detector system that responds to neutrons in the same way the human body does. Such an instrument, sometimes called a "neutron dosimeter," more commonly is called a "rem meter."

One major advantage of a BF_3 proportional counter is its lack of response to other radiations. Alpha and beta radiation cannot get through the walls of the tube or counter, and the small pulses produced by gamma rays can be discriminated out electronically. Thus, a BF_3 tube is capable of measuring neutrons in high gamma fields, typically to 500 R/hr. This capability can be an asset for health physics and for neutron detection in a reactor.

BORON-LINED COUNTERS

Another class of proportional counters used for neutron detection is described as boron-lined counters. In these detectors, thin layers of boron are used to coat inner walls of proportional counters. The boron permits a fill gas, such as methane-argon, to be used.

Even though this gas displays poorer qualities (than BF_3 gas) for use in proportional counters, many advantages of BF_3 are retained. And operation of a boron-lined counter is similar to that of a BF_3 counter. Operating voltage is usually lower - generally below 1000 volts. Also, plateaus are not flat. For these reasons, boron-lined counters are generally preferred to BF_3 counters.

The boron-lined proportional counter is extremely insensitive to gamma radiation, even to exposure rates of 10^4 to 10^5 R/hr. This type of detector is more likely to be used for in-core measurements in a reactor, or where high gamma fields exist. Comparably, a boron-lined counter has an order-of-magnitude less sensitive than a BF_3 proportional counter. But, for in-core reactor measurements, this range is no drawback, and may even be an asset because neutron fluxes may "swamp" (overcome) the resolving capability of a BF_3 detector. Another clear advantage of a boron-lined proportional counter for high-level work is its generally shorter resolving time, which permits it to measure neutron levels up to an order-of-magnitude greater than a comparable BF_3 detector.

^{3}He COUNTERS

Proportional counters filled with ^{3}He comprise a third general class of neutron proportional counters. These counters make use of the following reaction:

$$^3_2He + ^1_0n \rightarrow ^1H + ^3_1H + 765 \text{ keV}$$

with 191 keV going to the tritium nucleus and 574 keV to the proton.

The probability for this reaction is significantly greater than that of the $^{10}B(n,\alpha)^7Li$ reaction; however, the relatively low proton and triton energy produced gives rise to a relatively small pulse, and makes gamma rejection poor. Also, because of wide energy differences between the two particles, a poor plateau is obtained, and this problem is compounded by the poor stopping power of ^{3}He gas.

A ^{3}He-filled proportional counter has one very important application, however: neutron spectroscopy. The ^{3}He counter, because of its high sensitivity and very short pulse duration, is especially adaptable to use with "choppers" - devices that pass only certain energies of neutrons. Chopper action is very swift, and the fast response of ^{3}He chamber (because of its small pulse duration) makes it useful for this type of application.

The ^{3}He chamber can also be used for neutron "time-of-flight" studies - neutron energies are determined by noting or measuring the time it takes for neutrons to travel known path lengths. Again, rapid response and high sensitivity enable this application.

Note that ^{3}He chambers, though widely used, are found most commonly in research applications. When used for spectroscopy, they are designed with very high pressures, perhaps as great as 100 atmospheres, and high voltages, sometimes in excess of 300 volts. This design provides increased sensitivity and very short pulse durations.

^{3}He proportional counters also enjoy wide use in industrial gaging applications in which neutrons are used.

FISSION COUNTERS

Fission counters (chambers) make use of fission reactions. A neutron fission chamber is simply a proportional counter with a thin layer of fissionable material coated onto its inner walls.

Many different fissionable materials may be used, including:

^{235}U.

^{238}U.

^{237}Np.

^{239}Pu.

^{232}Th.

Response to neutrons depends upon specific characteristics of the fissionable material used. For example, ^{238}U and ^{232}Th are useful for detection of high energy neutrons.

The fission process produces two heavy nuclei (fission fragments), each with a kinetic energy of about 165 MeV. Thus, very large pulses with very short durations are produced. The fission chamber may be used, therefore, for in-core reactor measurements - the large pulse size permits excellent discrimination of gamma radiation, sometimes even to as great as 10^8 R/hr. The short pulse duration produces a short resolving time that, when coupled with the low, inherent sensitivity of the device, permits measurement of very high neutron levels.

High efficiency, and consequent applications in smaller neutron fluxes, can be achieved by multiple plates or foils inside a proportional counter. Generally, fission chambers are only about 1% as efficient (i.e., as sensitive) as BF_3 tubes of comparable size.

Fission proportional counters normally operate at fairly low voltages (only a few hundred volts) because of their very large pulse sizes. Since the range (distance of travel) of a heavy fission fragment is small, active volume need not be very large. [Note: Fission ionization chambers are sometimes used, because fission fragment energy (200 MeV) is high enough to produce a fairly large current without amplification.]

PROTON RECOIL COUNTERS

The final class of neutron proportional counter to be discussed is the proton recoil counter. It makes use of n,p scattering reaction. (The probability for this reaction is relatively large.)

A neutron proportional counting chamber making use of the proton recoil reaction is usually filled with a hydrogenous gas, such as methane or ethylene. Two drawbacks, however, are associated with these chambers:

> The neutron gives up only a part of its energy to the proton recoil, hence a variety of pulse sizes is produced - there is no real plateau and, more importantly, response of the detector is a direct function of the proton energy spectrum produced.
>
> The relatively high gamma sensitivity of these detectors can be improved somewhat by judicious selection of fill gas. In some cases it may be desirable to increase this sensitivity to gamma radiation, and make a detector that responds as a dosimeter to both gamma rays and neutrons.

Proton recoil proportional counters can be made in a wide variety of geometries, and are usually used in research laboratories or other special situations. While a fill gas rich in hydrogen is commonly chosen, hydrogenous walls (e.g., polyethylene or paraffin) also have been employed, as have plates or wafers within the sensitive volume.

A special application of the proton recoil proportional counter is the Hurst counter, designed by G.S. Hurst and his coworkers at Oak Ridge National Laboratory. The Hurst counter is filled with methane (CH_4) gas and contains both a thin and thick paraffin radiator (Figure 11-3-10). It can be

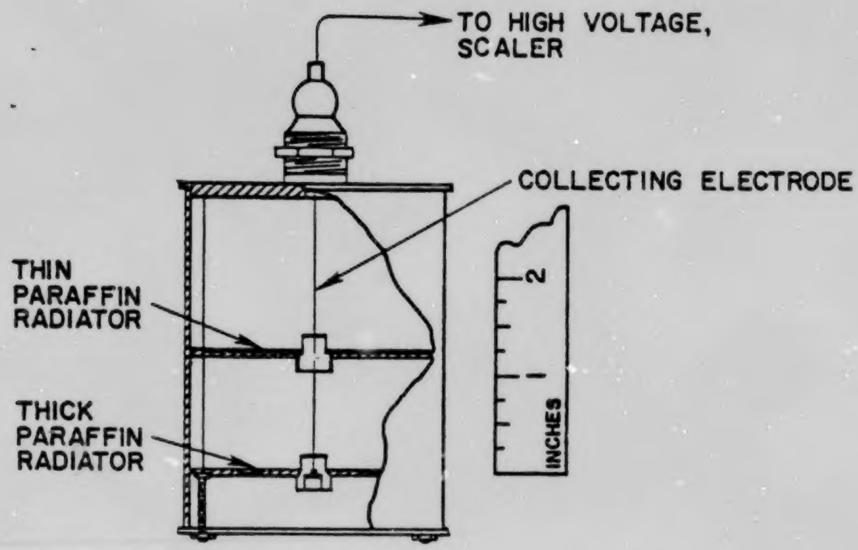
TO HIGH VOLTAGE, SCALER
COLLECTING ELECTRODE
THIN PARAFFIN RADIATOR
THICK PARAFFIN RADIATOR
2
1
INCHES

Figure 11-3-10. Fast neutron proportional counter tube, using proton recoil detection principle.

used to provide a relatively flat neutron response, regardless of neutron energy, over a range of neutron energies, from about 0.1 to 20 MeV. Other versions can be made to respond to dose rates over this neutron energy range.

OTHER APPLICATIONS OF PROPORTIONAL COUNTERS

A few special applications of proportional counters merit brief mention, including the Hurst dosimeter. This device employs a proportional counter chamber completely lined with polyethylene and filled with ethylene gas. Its chamber responds in the same way as human tissue to neutrons with energies from a few tenths MeV to 20 MeV. It can, therefore, be calibrated in units of rads/hr. The typical

Hurst dosimeter, often in the form of a portable survey meter, can be used for neutron dose rate measurement at levels as low as 1 mrad/hr.

Another application of proportional counters is for low-energy photon spectroscopy. A proportional counter tube, with a very thin window, is used for this purpose. The low-energy photons (x-rays) pass through this thin window with no significant energy loss, then undergo photoelectric interactions with the fill gas. Electrons produced give rise to gas amplification, and pulses are sorted according to size.

Fill gases commonly used for these x-ray proportional counters include such inert gases as Xe, Kr, Ar, or CO_2 at atmospheric pressure. Operating voltages are quite high - 2000 volts and up. If used with an ultra-stable power supply, such proportional counters provide excellent resolutions of low energy x-rays, far better than any other type of photon spectrometer.

Gas-filled proportional counters also may be used as beta spectrometers. As with the low-energy photon application, the high gain or multiplication obtained from gas amplification makes this application feasible. Inert gases are ofted used (with the exception of argon, which experiences long-lived excited states - from beta reactions - that cause after-pulsing). This after-pulsing problem, of particular significance in beta spectrometry, requires ultra pure gases; P-10 (90% argon, 10% methane) is commonly used.

Proportional counters also are used for alpha spectrometry and as flow-through counters for radioactive gas monitoring. These applications require a highly stable power supply, accurate to 0.1%. Any significant drift or change in voltage - even as little as 1% - significantly changes response of the instrument. For example, the gas-amplification factor shifts by as much as a factor of 2 with only a 10-volt change; for a counter operating at 2000 volts, this voltage change is only 0.5%.

In conclusion, it should be noted that the proportional counter is highly versatile; because of gas amplification, it permits measurements in situations in which ionization chambers are inadequate.

EXERCISES

1. What gas amplification factor is shown in Figure 11-3-2?
2. The resolving time of a proportional counter is 0.2 msec for β particles and 0.5 msec for α particles. What is the maximum counting rate theoretically possible for each? If the efficiency of the detector is 48%, how big a source could be counted (theoretically)? Suppose the beta source strength was 2 μCi. What would the resolving time loss be?
3. What would you expect to happen if a gas flow proportional counter had a pinhole leak in its window?

MATERIALS

1 proportional counter with variable discriminator and variable power supply.
1 alpha source.
1 mixed alpha-and-beta source.
1 beta source.
Linear graph paper.

PROCEDURES

1. Turn proportional counter on; place alpha source into counting position; increase voltage slowly until unit begins to count; note voltage.
2. Step up voltage slightly to a convenient reference point; begin timed count for a few minutes to obtain good counting statistics; record counting data on Data Table 11-3-1.
3. Step up voltage by 50 volts; take another count; record the data.
4. Continue taking counts until readings are no longer on plateau; record all data.
5. Repeat experiment (1-4, above) with beta source. For both series of counts, discriminator should be set near zero.
6. Plot count rate versus voltage; determine best operating voltage for tube; with the instrument set at this voltage, begin count with alpha-and-beta source in place; determine combined source strength.

7. Increase discriminator setting until betas are no longer being counted - only alphas; determine alpha source strength; increase voltage to see if a beta counting plateau can be determined.

DATA TABLE

DATA TABLE 11-3-1
PROPORTIONAL COUNTING

VOLTAGE	COUNTS	TIME	COUNTS PER MINUTE

Discriminator setting ______________________________

Source used ____

Operating voltage: α source _______________________

β source _______________________

REFERENCE MATERIALS

Chase, G.D. and Rabinowitz, J.S. Principles of Radioisotope Methodology. Minneapolis, MN: Burgess Publishing Co., 1965.

"Instrumentation for Environmental Monitoring." Lawrence Berkeley Laboratory Report, LBL-1 3 (October 1973).

Korff, S.A. Electron and Nuclear Counters. 3rd ed. New York: D. Van Nostrand, 1968.

Maushart, R. Radiation Protection Instrumentation. Oxford: Pergamon Press, 1972.

Price, W.J. Nuclear Radiation Detection. 2nd ed. New York: McGraw-Hill, 1965.

"Radiation Protection Instrumentation and Its Application." International Commission on Radiation Units and Measurement Report, No. 20, 1971.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. Draw and label a simple block diagram of a proportional counter circuit.

2. Briefly describe gas amplification and explain its role in proportional counting.

3. List four factors that permit the amount of gas amplification to be varied (increased or decreased) in proportional counters.

   a.

   b.

   c.

   d.

4. Which type of detector, ionization chamber or proportional counter, is more sensitive? Why? ______________________________

____

____

5. Briefly describe and differentiate among these kinds of proportional counters: internal, 4π, and flow through.

   a. Internal: ______________________________

   ____

   ____

   b. 4π: ______________________________

   ____

   ____

   c. Flow Through: ______________________________

   ____

   ____

6. What is the purpose of the discriminator? ______________________________

____

____

____

7. Why would an internal proportional counter be an ideal choice for absolute or precision counting? ______________________________

____

____

____

8. Calculate the percent of counts lost, given a counting rate of 800,000 cpm and a resolving time of 1 microsecond.

9. List the five basic types of neutron proportional counters.

   a.

   b.

   c.

   d.

   e.

10. Explain why a BF_3 (boron tri-fluoride) counter is relatively good for making neutron measurements in a mixed neutron and gamma radiation field.

11. Of what main elements is P-10 gas comprised? Why is P-10 a popular gas to use in proportional counters?

12. What is resolving time? Why is it important to have a small resolving time?

13. Why are fission counters well suited to the measurement of very high neutron levels?

14. What are two major drawbacks associated with neutron proportional chambers that make use of the proton recoil reaction?

15. Briefly explain the purpose and operation of the Hurst counter.

16. Briefly explain the purpose and operation of the Hurst dosimeter.

17. The following data was obtained from an internal proportional counter with a radiation source emitting Alpha and Beta particles.

Applied Voltage (Volts)	Detector Count Rate (cpm)
800	0
900	4,200
1,000	5,300
1,100	5,350
1,200	5,400
1,300	5,450
1,400	7,500
1,500	11,000
1,600	15,500
1,700	18,250
1,800	19,000
1,900	20,000
2,000	22,000

Do the following:

a. Plot a plateau curve from this data.
b. Identify the Alpha plateau and Beta plateau.

18. Explain the discriminator effects which result in the two plateaus for the Alpha and Beta radiation. ______________________________

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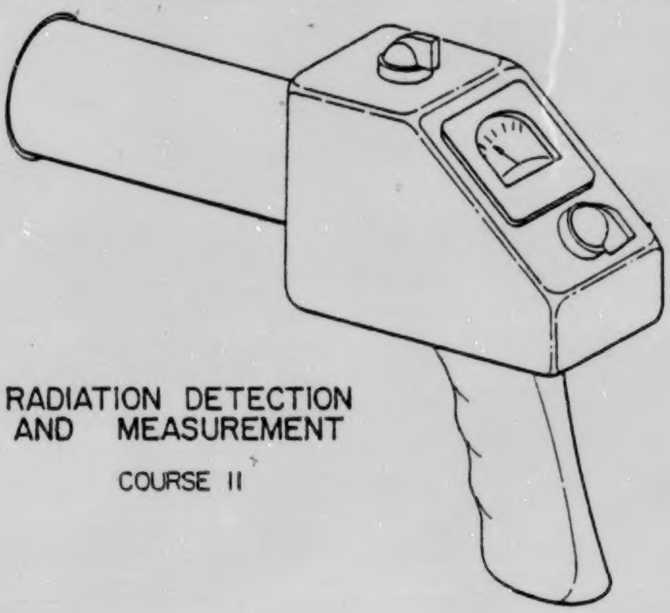
NUCLEAR TECHNOLOGY

MODULE 11-4 GAS-FILLED DETECTORS III: GEIGER-MUELLER COUNTERS.



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
⇨ MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

Psychologists often use word association tests when evaluating people. In a word association test, the psychologist says a word, and the patient responds with the first thing that pops into the patient's mind. For example, say the word "big" to someone, and the response is likely to be "small" or "little." Similarly, say "dog" and the likely response is "cat." If someone says the word "radiation," "Geiger counter" might well be the instinctive response.

Almost everyone has heard of Geiger counters and, in the minds of most people, this counter symbolizes the Atomic Age. All instruments used in the nuclear field are, in the minds of the public, Geiger counters; it is this term that has found its way into our everyday language.

The Geiger counter however, is a very specific kind of radiation detector, among the oldest of gas-filled counters. As early as 1907, Hans Geiger, a German scientist working in England with the great Ernest Rutherford, used gas-filled tubes to count alpha particles; these tubes were forerunners of the modern Geiger counter.

In 1913, Geiger first made what might be correctly called a Geiger counter. In 1928, working in Germany with Walther Mueller, he developed a much refined and improved version that took advantage of the great advances in electronics brought about (in the 1920's) by invention of the vacuum tube. That 1928 instrument is essentially identical to the modern version, which is better known as a Geiger-

Mueller (GM) counter.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Briefly describe the theory of operation of a Geiger-Mueller counter.
2. Draw a GM plateau and label all relevant parts.
3. Calculate the following for a GM detector:
   Resolution time.
   Geometry factor.
   % efficiency.
   % slope.
4. Differentiate between internal and external quenching.
5. Explain the following terms:
   Limited proportionality.
   Breakdown voltage.
   Continuous-discharge region.

Multiple discharge.
Quenching.
Dead time.
Recovery time (resolving time).
Geometry factor.
Self-absorption.

6. Given the appropriate equipment:
Determine GM plateau and operating voltage.
Obtain proper background rate.
Determine resolution losses and resolving time.
Determine GM tube efficiency for a variety of alpha, beta and gamma sources.

THEORY AND PRINCIPLES OF OPERATION

If the voltage on a detector counting in the proportional region (Region III, Figure 11-4-1) is increased, pulse sizes (heights) will continue to grow larger. But, at some point, the size difference between the pulse produced by an alpha particle (highly ionizing) and a beta particle begins to get smaller, until both the alpha particle and beta particle are producing pulses of the same size. In other words, a point is reached at which it is not possible to discriminate electronically between alpha and beta particles because pulses are of the same size.

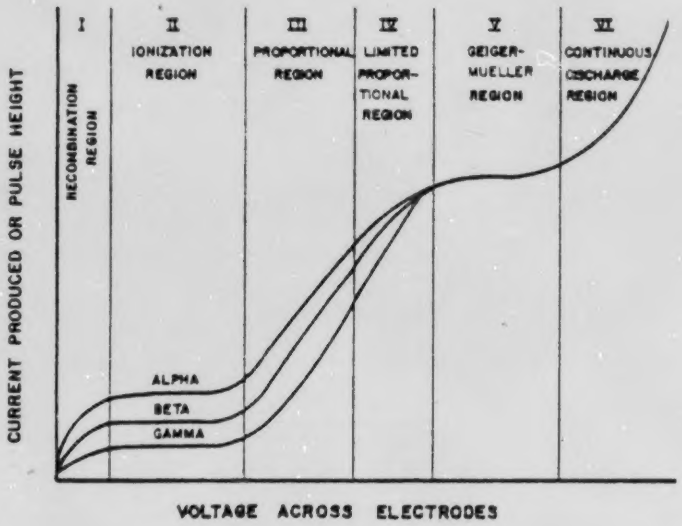

Figure 11-4-1. Idealized curve: current output versus voltage of a gas-filled detector for highly ionizing radiations (gamma).

NOTE: In Region III, slopes for the three radiations remain the same; in Region IV, curves begin to converge, finally joining at the GM threshold.

This region of limited proportionality is labeled Region IV. In it, even gamma rays, relatively weakly ionizing, produce a pulse about the same size as the pulse produced by an alpha particle. Also, in this region, pulse height is not proportional to energy of an incoming particle, as it is in the proportional region (III).

The region of limited proportionality is not used for radiation detection and measurement. It has no real advantage over Region III. And, it does not permit discriminations, since all radiations produce pulses of approximately the same size. Furthermore, larger and more expensive power supplies are needed to produce the higher voltages required for operation in this region - higher voltages reduce the life of a proportional detector.

Even though Region IV has no application to radiation detection or measurement, it should be noted that, at the upper end, all ionizing events in the detector produce the same size pulse. In other words, the number of ions or current flow is the same for one alpha particle, which may produce more than 100,000 ion-pairs, as for one gamma ray, which may produce only 10 or 15 ion-pairs. Thus, there is an interesting state of affairs, in which the gas-amplification factor is different for each type of radiation. But the product of this factor and the number of ions produced by the initial ionizing event (I) are constant.

The point at which this phenomenon occurs is actually the starting point or threshold of the Geiger-Mueller region (V).

The reason for this effect is electronic, and can be rather simply explained. In a proportional counter or ion chamber, the gas "avalanche" is produced or collected only at one point along the collecting electrode. Because of a higher voltage in the Geiger-Mueller region, however, the avalanche spreads along the entire length of the collecting electrode. Thus, a very large pulse is produced and, as was shown, the size or height of this pulse is independent of the number of ion pairs that produced it.

This feature or property leads to a highly sensitive counter. On the other hand, it also produces certain limitations not associated with other gas-filled detectors (ion chambers and proportional counters).

(Note: the curves shown in Figure 11-4-1 are illustrations only; it is quite difficult to construct a detector that operates effectively in all regions.)

If voltage on the detector is further increased, pulse size levels off, and a plateau is reached (Figure 11-4-2).

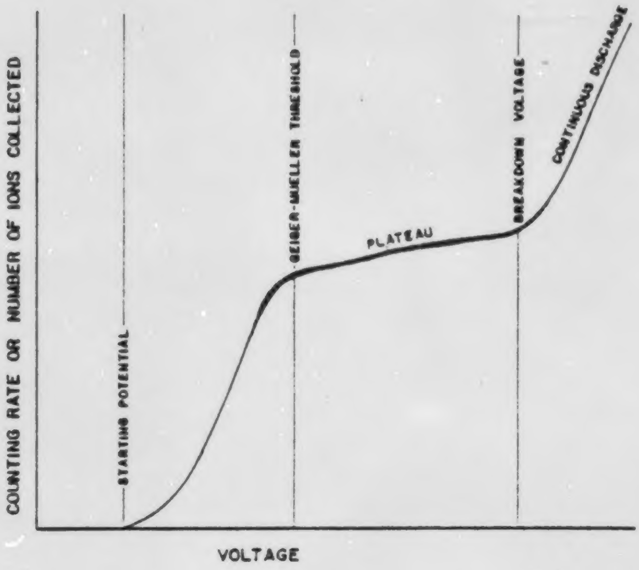

Figure 11-4-2. Typical curve for detector operating in Geiger-Mueller region.

This Geiger plateau, which has only a slight slope upward, may extend for a few hundred volts. Then, if voltage is increased further (to the breakdown voltage), arcing occurs across the electrodes, even in the presence of no radiation. The tube conducts continuously or is in continuous discharge. Caution: Operation in this region quickly damages tube.

The GM counter is designed so that pulses in the ionization and proportional regions are too small to be counted.

A typical GM circuit (Figure 11-4-3) registers a count each time a voltage pulse reaches the counter. A variable high-voltage supply is used to obtain proper operating voltage. The cathode is at ground (zero) potential whenever a positive charge is placed on the collecting electrode.

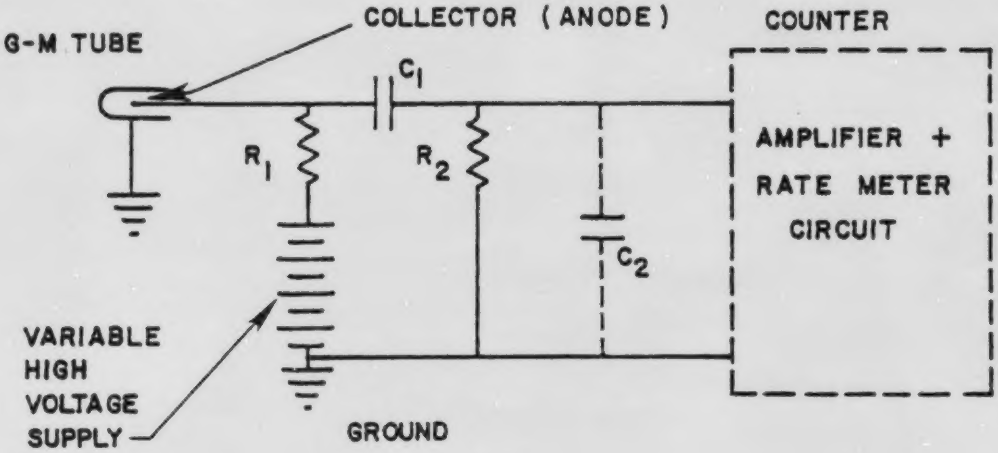
G-M TUBE
COLLECTOR (ANODE)
COUNTER
C_1
R_1
R_2
AMPLIFIER + RATE METER CIRCUIT
C_2
VARIABLE HIGH VOLTAGE SUPPLY
GROUND

Figure 11-4-3. Simplified schematic diagram of Geiger-Mueller counting circuit.

NOTE: C_2 refers to total counter capacitance.
R_2 refers to resistance of counter circuit.

Capacitor C_1 blocks high voltage from the counter, while resistor R_1 permits voltage (on collector electrode) to drop when negative ions are collected. The ions are collected by isolating or separating the high-voltage supply from the collecting electrode. Typically, R_1 is a very large-valued resistance, on the order of 1 or more megohms.

Response time, expressed as the "time constant," a product of resistance (R_2) and capacitance (C_2), is a measure of how quickly the instrument responds. This time constant has a logarithmic derivation that is quite complex. For the purpose of this module, in one time constant, the instrument responds 63% of the maximum (full-scale) reading (Figure 11-4-4) and, in two constants, 86.5% of the maximum. This time constant, which is of particular importance in the operation of a portable survey meter, whether GM or some other type, is a function of the electronic circuitry used.

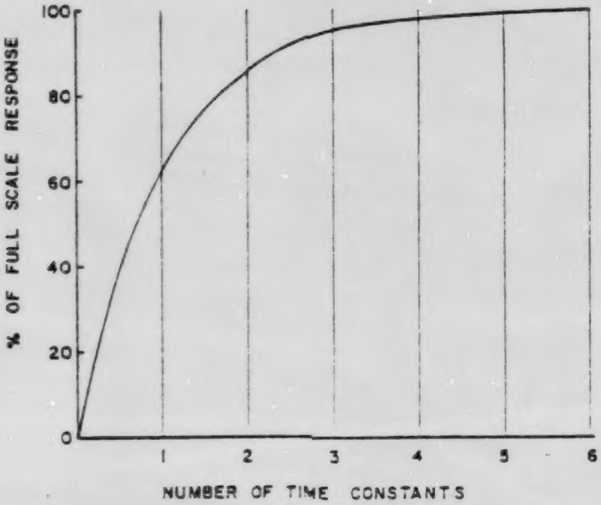

Figure 11-4-4. Relationship between fraction of full-scale response and time constant.

As mentioned previously, primary ionization in the GM tube produces, through the avalanche, about 106 ion pairs. The electrons, being small and mobile, are quickly collected by the anode (in most counters, a thin wire running down the middle of a cylindrical tube). Left behind are the much larger and slower positive ions.

As long as the positive ions remain, their presence produces undesirable effects in the GM tube. For example, when positive ions reach the walls (cathode) of the tube (usually at ground or zero potential), they cause electrons to be liberated, and these can produce a new avalanche. Thus, the initial avalanche may be followed by many other avalanches, and a whole series of discharges may occur at different intervals (known as multiple discharge). These discharges, in some instances, can appear continuous and ruin the ability of the GM counter to count individual ionizing events.

Multiple discharges, which occur primarily in counters containing pure gases, result from positive ions migrating all the way to the cathode where they combine with electrons in the metal. A process known as quenching prevents these secondary electrons from reaching the collecting electrode and thereby eliminates multiple pulsing. Quenching can be accomplished internally or externally.

External quenching is accomplished electronically, by means of a high resistance, or by a special electronic circuit that renders the tube dead for a period of time after an initial pulse is received. This circuit also acts to effectively lower the amount of time the tube is available for counting; that is, it makes the resolving time very long, on the order of 1 to several milliseconds. This arrangement severely limits the detector's upper counting range.

Use of a high-valued resistor offers the disadvantage of increasing the circuit's time constant.

Such tubes are seldom used and are not often offered for sale commercially.

Internal quenching, also known as self-quenching, is widely used. Self-quenched tubes contain a small amount (or percentage) of a quenching gas in the main filling gas. Two types of quenching gas, which ionizes more readily than the counting gas, are in common use:

Organic gas, such as methane, ethane, ethanol (ethyl alcohol) and ethyl formate.

Halogens, such as bromine or chlorine.

A quenching gas, basically, serves as an electron transfer device. In a GM tube, the major gas constituent is the counting gas (for example argon), which is ionized by the radiation. Electrons produced from this ionizing process migrate to the anode, leaving behind positively charged ions (mostly argon ions). The migrating ions collide with the quenching gas (e.g., ethanol), neutralizing argon ions by transfer of one (or more, as required) electrons from ethanol. The ethanol, now positively charged, migrates toward the cathode where it is neutralized. Meanwhile, the energy released does not go into the production of secondary electrons; rather, it goes to dissociate the ethanol molecules.

Because the quenching gas is being dissociated or destroyed in the process, a point is reached when there is no longer enough quenching gas to prevent multiple discharges. With organically quenched tubes, this point is in the neighborhood of 10^{10} ionizing events or counts; halogen quenched tubes seem to have unlimited lifetimes because halogen gas dissociation reaction is reversible - halogen gas atoms recombine, that is reform molecules of halogen gas.

Despite their finite lifetime, organic-quenched tubes have the advantage of a long, gently sloping plateau. Such plateaus are desirable because they permit lower voltage operation, and experience little change in counting rate for slight voltage changes. Halogen-quenched tubes typically have shorter plateaus with steeper slopes.

Dead time is that period immediately after a pulse in which the instrument remains insensitive and unable to record another ionizing event. Because of the need for quenching, dead time is relatively long, typically in the neighborhood of 100 to 200 μsecond. The actual duration of a voltage pulse in the GM tube is short - perhaps 20 μsecond - but time is needed for the positive ions to recombine and for the voltage.on the anode (collector) to return to a point at which another pulse can be collected (Figure 11-4-5).

Recovery time is the minimum time that must elapse between two events to permit their being recorded as individual events. During this recovery time, voltage on the anode again rises and, although capable of producing a pulse, its pulse may be smaller than normal. No pulse is counted unless it is large enough to pass the discriminator or to drive the counting circuit. By definition, pulses formed during the recovery period are of insufficient size to register a count.

Resolution time in a GM tube necessitates corrections for pulses occuring during the dead time. Losses of counts, which can be excessive at high counting rates, can be corrected mathematically:

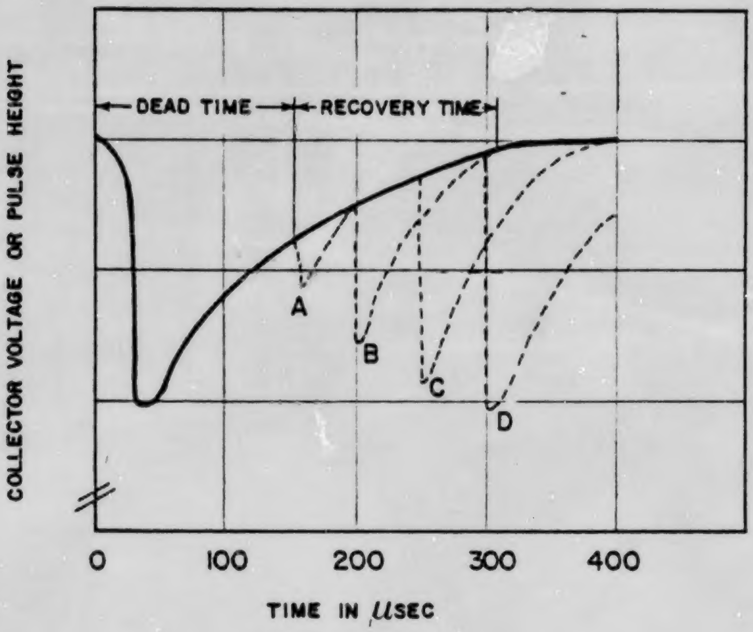

Figure 11-4-5. Relationship of dead time and recovery time in a Geiger-Mueller tube. Ionizing event took place at zero time; the large drop in voltage occurred when negative ions (from avalanche) reached collector. During dead time, the tube was being quenched. A, B, C and D show pulse sizes and shapes from ionizing events during recovery time, at 150, 200, 250 and 300 μsecond respectively after initial pulse. Full recovery is achieved 300 μsecond after initial pulse, but tube can respond again 150 μsecond later. Hence, the resolving time is 150 μsecond.

$$R = \frac{R_{obs}}{(1 - R_{obs} T)}$$

Equation 11-4-1
(Equation = Eq.)

where

R = true or correct counting rate (i.e., rate that would be obtained if there were no dead time).

R_{obs} = observed counting rate.

T = dead time.

NOTE: Any convenient time base (e.g., minutes or seconds) can be used, but time base must be the same for all terms in equation.

EXAMPLE

Given:

Counter with resolving time of 500 µsec (= 500 x 10^{-6} sec); observed count rate of 5000 cpm.

Find: True count rate.

Solution: (Use Eq. 11-4-1).

$$R = \frac{R_{obs}}{(1 - R_{obs}\ T)} .$$

Substituting,

$$R = \frac{5000\ \frac{\text{counts}}{\text{minute}}}{\left(1 - 5000\ \frac{\text{counts}}{\text{minute}} \times 500 \times 10^{-6}\ \text{second}\right)}$$

$$= \frac{5000\ \frac{\text{counts}}{\text{minute}}}{\left(1 - \frac{5000 \times 500 \times 10^{-6}\ \text{counts} \times \text{sec}}{1\ \text{min.}} \times \frac{1\ \text{min.}}{60\ \text{sec.}}\right.}$$

$$= \frac{5000\ \frac{\text{counts}}{\text{minute}}}{1 - 0.04} = \frac{5000\ \frac{\text{counts}}{\text{minute}}}{0.96}$$

$$= 5020\ \frac{\text{counts}}{\text{minute}} .$$

DESIGN AND APPLICATIONS

Geiger-Mueller tubes are made in a variety of sizes and shapes, suitable for use in laboratory counting situations as pulse counters for alpha, beta and gamma radiations and in portable radiation monitoring instruments (Figure 11-4-6). Table 11-4-1 describes some of the features of these tubes.

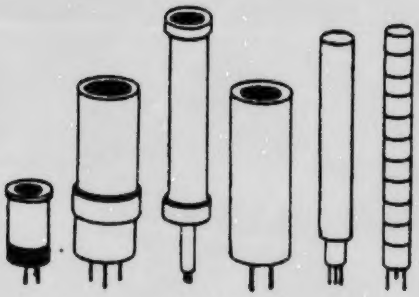

Figure 11-4-6. Common geometries for Geiger-Mueller tubes. Four tubes on left have thin end-windows, and fairly thick walls. Two tubes on right have no windows, and uniform wall thicknesses. Ribs on tube (far right) give strength; its stainless steel wall is 30 mg/cm^2 thick. Note different bases, ranging from 1 to 4 pins.

For most laboratory counting situations, thin end-window GM tubes are used. Wall thicknesses are expressed in mg/cm^2, and a thickness of 30 mg/cm^2 or less is sufficient for most beta counting operations. Walls with this thickness can be easily made from aluminum or stainless steel.

TABLE 11-4-1
CHARACTERISTICS OF SOME COMMON GEIGER-MUELLER TUBES

DESCRIPTION	WINDOW OR WALL (mg/cm^2)	OPERATING VOLTAGE	PLATEAU LENGTH	PLATEAU SLOPE	CATHODE MATERIAL	DIMENSION
Thick mica window halogen quench tube	3 - 4	900 V	200 V	10%	Stainless Steel	1 3/8" dia x 4 11/32" long
Thin mica end window	1.4 - 2	900 V	200 V	5 - 10%	Stainless Steel	1 3/8" dia x 4 11/32" long
Thin mica end window halogen quench	1.4 - 2	900 V	180 V	10%	Stainless Steel	1" dia x 6" long
General purpose detector	35	900 V	150 V	3%	Carbon	21/32" dia x 5 1/4" long
Mica end window small tube	1.5 - 2	600 V	250	2%	Stainless Steel	11/32" dia x 2 1/4" long
Pancake tube	1.2	950 V	200 V	5 - 10%	Stainless Steel	2 1/8" dia x 3 5/32" long
Large volume x-ray detector	1.2	1175 V	200 V	5%	Stainless Steel	1" dia x 6" long
Medical-isotope Tracer studies	30	700 V	150 V	3 - 5%	Stainless Steel	9/16" dia x 8.6" long

For increased sensitivity, or for counting low-energy beta or alpha particles, a tube with a mica window of a few mg/cm^2 might be used.

Mica window tubes can be made with window thicknesses down to about 0.4 mg/cm^2, but about 1 mg/cm^2 is the practical minimum. Note: The expression mg/cm^2 is useful in that it provides independence from the density of the wall material; to obtain this thickness, it is only necessary to divide by the density of the wall material.

EXAMPLE

Given:

An aluminum window with a thickness of 20 mg/cm^2.

Find: Thickness in centimeters and inches.

Solution: Density (aluminum = 2.7 g/cm³).

$$\text{Thickness (cm)} = \frac{20\ \frac{\text{mg}}{\text{cm}^2}}{2.7\ \frac{\text{g}}{\text{cm}^3}} = 7.4\ \frac{\text{mg}}{\text{g}}\ \text{cm}.$$

Since,

$$1\ \text{mg} = 0.001\ \text{g}$$

$$\text{Thickness (cm)} = 7.4\ \frac{0.001\ \text{g cm}}{\text{g}}$$

$$= 0.0074\ \text{cm}.$$

$$\text{Thickness (in)} = 0.0074\ \text{cm} \times \frac{1\ \text{in}}{2.54\ \text{cm}}$$

$$= 0.0029\ \text{inches}$$

$$= 2.9\ \text{mils}.$$

When a GM counting system is set up in a laboratory, its operating characteristics must be determined. First, a plateau is "run," and from this data, a operating voltage can be determined. (The operating voltage selected is normally one-third of the way up the plateau.) This voltage minimizes chances of tube damage from over wattage, and it prolongs tube life. Plateau slope is also determined (Figure 11-4-7) as follows:

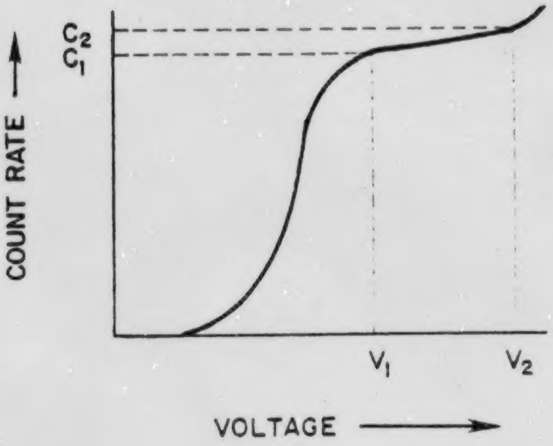

Figure 11-4-7. Illustration of % slope calculation.

$$\% \text{ slope} = \frac{(C_2 - C_1)}{0.01\,(V_2 - V_1)} \times 100 \qquad \text{Eq. 11-4-2}$$

where

% slope = percentage change in count rate per hundred-volt increment.
C_2 = count rate at voltage V_2.
C_1 = count rate at voltage V_1.

Resolving time also must be determined. In this determination (described in laboratory Exercise 2) a count is taken with one source, then with another, then with both. Resolving time τ may then be calculated as follows:

$$\tau = \frac{2(R_1 + R_2 - R_{12})}{(R_1 + R_2)\ R_{12}} \qquad \text{Eq. 11-4-3}$$

where

R_1 = count rate from source 1.
R_2 = count rate from source 2.
R_{12} = count rate from counting two sources simultaneously.

EXAMPLE

Given:

An exercise to obtain resolving time was carried out in the laboratory with these results:

R_1 = 3124 cpm, first source counted by itself.
R_2 = 2876 cpm, record source counted by itself.
R_{12} = 5980 cpm, both sources counted together.

Find: Resolving time.

Solution:

Use Equation 11-4-3.

$$T = \frac{R_1 + R_2 - R_{12}}{2\ (R_1 R_2)} .$$

Substituting,

$$T = \frac{3124 \text{ cpm} + 2876 \text{ cpm} - 5980 \text{ cpm}}{2\ (3124 \text{ cpm} \times 2876 \text{ cpm})}$$

$$= \frac{20 \text{ cpm}}{17.9 \times 10^6\ (\text{cpm})^2}$$

$$= 1.12 \times 10^{-6} \text{ minute} \times \frac{60 \text{ sec}}{\text{minute}}$$

$$= 6.7 \times 10^{-5} \text{ sec.}$$

Efficiency of a detector is determined by counting a source of known strength and calculating the fraction or percent of the disintegrations that are recorded as counts:

$$\%\ \text{efficiency} = \frac{\text{observed counting rate}}{\text{known disintegration rate}} .$$

Eq. 11-4-4

EXAMPLE

Given:

A calibrated ^{137}Cs source with an activity of 8.7 µCi; a net count rate of 876 cpm.

Find: Efficiency (% Eff) of counting system.

Solution: Use Equation 11-4-4.

$$\% \text{ eff} = \frac{\text{observed counting rate}}{\text{known disintegration rate}}$$

$$= \frac{876 \; \frac{\text{counts}}{\text{minute}}}{8.7 \times 10^{-6} \text{ Ci} \; \frac{3.7 \times 10^{10} \frac{\text{dis}}{\text{sec}}}{\text{Ci}} \; \frac{60 \text{ sec}}{\text{min}}} \times 100\%$$

$$= \frac{876 \times 100\%}{1931.4 \times 10^{4}} , \frac{\text{counts}}{\text{disintegration}}$$

$$= 0.0045\%, \frac{\text{counts}}{\text{disintegration}} \; .$$

The efficiency of a GM detector (or any detector) is dependent upon many factors, the most important of which is its geometry factor (G). This factor is the fraction of particles (from the source) that start toward the sensitive volume (window) of the detector (Figure 11-4-8).

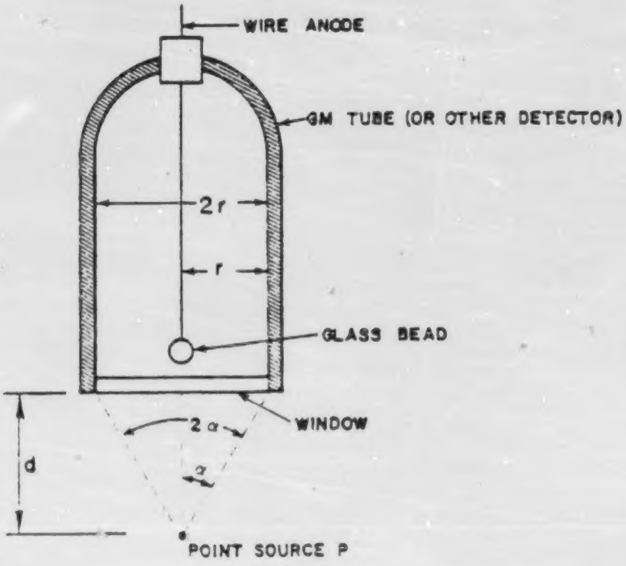

Figure 11-4-8.
Relationship of source to detector geometry.

Only those particles emitted within the solid angle, represented by 2 α, enter the tube. Geometry factor G can be calculated for a circular window detector (for a point source, or if source is very small compared with detector size):

$$G = 0.5 \left(1 - \frac{d}{\sqrt{d^2 + r^2}}\right). \qquad \text{Eq. 11-4-5}$$

EXAMPLE

Given:
A small radiation located in a standard sample holder 2.5 inches from window of GM tube; tube radius is 0.5 inches.

Find: Geometry factor.

Solution: Use Equation 11-4-5.

$$G = 0.5 \left(1 - \frac{d}{\sqrt{d^2 + r^2}}\right)$$

Substituting,

$$G = 0.5 \left(1 - \frac{2.5}{\sqrt{2.5^2 + 0.5^2}}\right)$$

$$= 0.5 \left(1 - \frac{2.5}{6.5}\right)$$

$$= 0.5\ (1 - 0.385)$$

$$= 0.5\ (0.615)$$

$$= 0.308.$$

In this equation, d is the distance from the source to the detector, and r is the radius of the detector. The greater the distance, the smaller the fraction reaching the window, and the smaller the geometry factor. Similarly, the larger the window, the larger the value of G. [Note: Equations 11-4-2 through 11-4-5 apply to all types of pulse counters (ionization chambers and proportional counters) and GM's.]

In addition to the geometry factor, several other factors influence detector counting efficiency:

Self-absorption (the absorption of the particles within the source itself).

Absorption (attenuation of the particles by air or the detector window).

Probability of producing a pulse, once a particle enters chamber.

Absorption is significant with alpha emitters and to a lesser extent with beta emitters, particularly low-energy beta-emitters. But for all practical purposes, any alpha or beta particle that gets into the chamber produces a pulse, assuming no dead time losses. This likelihood does not exist for gamma radiation however, because of its low probability of interaction with fill gas (in counter).

Geiger-Mueller tubes, widely used in portable survey meters and other instruments designed for radiation monitoring in the field, are miniaturized electronics packages with rate meters that use GM tubes as a detector (Figure 11-4-9):

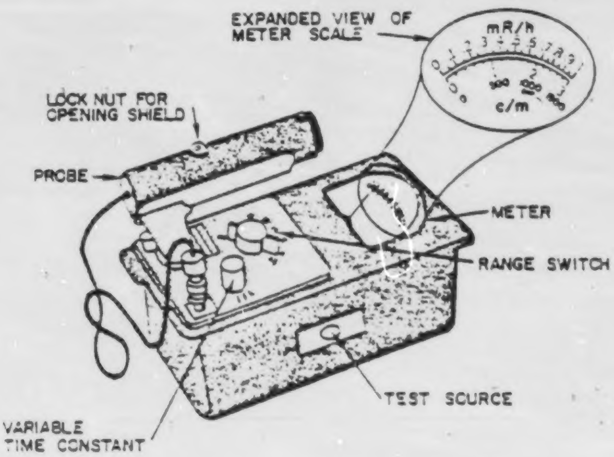
EXPANDED VIEW OF METER SCALE
mR/h
c/m
LOCK NUT FOR OPENING SHIELD
PROBE
METER
RANGE SWITCH
TEST SOURCE
VARIABLE TIME CONSTANT

Figure 11-4-9. Portable Geiger-Mueller survey meter (Victoreen Instrument Company Model 491).

For making beta-gamma surveys, GM survey meters equipped with stainless-steel walled tubes (30 mg/cm^2 thick), about 10 cm long and 2 cm in diameter are used. In such a meter the tube is encased in a housing (probe) with a metal window that can be opened or closed by rotating the housing around its long axis. In the window-open position, the exposed portion of the tube responds to both beta and gamma radiation. With the window closed, the tube is sensitive to gamma radiation only. The shield around the tube also absorbs (attenuates) low-energy gamma radiation.

Most GM survey meters use halogen-quenched tubes designed to operate at about 900 volts. This design greatly simplifies the electronic circuitry - no variable power supply is required. Many commercially made, portable GM survey meters obtain their GM tube high voltage from a small, inexpensive, gaseous voltage-regulator tube.

For monitoring alpha and low-energy beta particles, a flat, round tube (known as a "pancake tube") with a 1.2 mg/cm^2 window is widely used. (When used in the housing of a portable instrument, the unit is called a "pancake probe.") These tubes are sensitive to high-energy betas as well as gamma and x-rays, but they are too expensive and too fragile for such use because of their thin window than their cylindrical stainless-steel tube construction.

Response for a GM counter is not proportional to energy deposited in the chamber; therefore, it is not a dose or exposure rate meter. But, because the dose-per-unit of gamma energy flux is remarkably constant over the energy range of about 60 keV to 1.5 MeV, it is possible to calibrate a GM detector in exposure units. Also, because most GM tubes have high Z walls and fill gases, compared with the Z of tissue or air, they have a high energy dependence

(Figure 11-4-10).

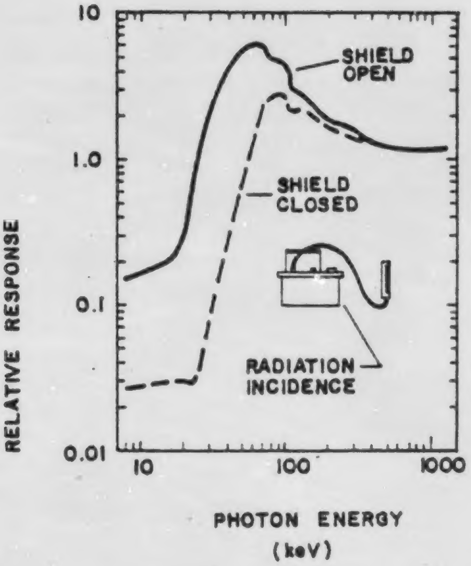

Figure 11-4-10. Energy dependence curve for Geiger-Mueller survey meter. Co-60 energies are 1250 keV; fission produce gammas are about 700 keV.

Another survey meter drawback is its characteristic of saturating (called "paralysis") in high level radiation fields. Characteristically, some GM survey meters move downscale in relatively high radiation fields, reading zero at some point. This characteristic could be hazardous if, for example, a field were actually 100 times or more higher than indicated on the meter.

To a great extent, the saturation problem can be corrected by proper circuit design at slightly increased cost. GM instruments can be made that saturate (i.e., read full up-scale) at levels 1000 times or more their full up-scale values (perhaps up to a few hundred R/hr). Others move down-scale at levels as low as a few R/hr, or less.

Because of the long resolving time of a GM tube, most GM survey meters are limited to full-scale readings of between 50,000 and 100,000 k counts per minute, which is equivalent to 20 to 50 mR/hr of gamma radiation.

Some GM instruments are made to change operating modes above a certain level, becoming mean-level current detectors instead of pulse counters. These instruments usually are equipped with tiny GM counters (or, in some instances, ion chambers) inside their electronics cases. Such a detector is far less sensitive than the external-probe type because of its size and design. But, it can extend the range of a GM survey meter tremendously; at least one type of commercial GM instrument measures gamma fields up to 1000 R/hr. This instrument is equipped with a telescoping probe that permits the operator to measure a radiation field 15 feet away; its probe also can be safely immersed in water.

As an aid to low-level monitoring, portable GM survey meters can be equipped with speakers or earphones that click each time a pulse is recorded in the counting circuit. Thus, the operator has an aural indication of changing (or increasing) radiation fields. These changes may be more easily and quickly discernable aurally than from the meter, which tends to fluctuate or "bobble" because of the statistical nature of radioactivity.

Fluctuations are, of course, more pronounced at low levels than at high levels, in accordance with the principles put forth in Module 11-1, "Statistics of Radiation Counting."

EXERCISES

1. An organic-quenched GM tube has a rated lifetime of 2×10^{10} counts. How long would it last if it were used 20 hours a day, five days a week, to count air samples at an average counting rate of 100 counts per minute?
2. Make a chart of observed versus true counting rates; use Equation 11-4-3 for detectors with resolution times of 100, 150, 200 and 250 μsec. (Plot four curves on one chart.)
3. Determine thickness in centimeters and mils for a GM tube with 30 mg/cm^2 stainless-steel walls. (The density of stainless steel is 7.85 g/cm^3.)

MATERIALS

1	Laboratory scaler with end-window GM detector in sample holder or counting pig.
1	Gamma source of known strength.
1	Alpha source of known strength.
	Linear graph paper (Cartesian coordinates).
2	Beta sources of approximately equal strength (a "split source" is preferred).
3 or more	pure beta sources of known strength. (Suggested sources: ^{14}C, ^{90}Sr - ^{90}Y, ^{204}Tl.)

GEIGER-MUELLER PLATEAU AND OPERATING VOLTAGE

1. Check scaler and detector to be sure they are properly connected; make sure high voltage is turned all the way down.
2. Insert source into counting pig; center it under window of GM tube.
3. Turn counter on; reset and take a one-to-two minute count; record data on Data Table 11-4-1.
4. Increase voltage gradually, in 50 volt increments; take count for 1 to 2 minutes at each increment; record data on data table.
5. Continue procedure through plateau until count rate rises very rapidly; terminate experiment at this point to avoid damage to tube.
6. Plot counting rate as a function of voltage on graph paper; use data from data table; label various parts of curve (threshold voltage, plateau, breakpoint voltage).
7. Complete Data Table 11-4-1.

DETERMINATION OF RESOLUTION TIME

1. Turn equipment on; allow it to warm up briefly.
2. With no source in place, and with sources remote from detector, allow instrument to count for at least 20 to 30 minutes, or until 1000 counts have accumulated - this count provides "background;" calculate back-

ground count rate in terms of counts per minute; [background must be subtracted from counts made with source to determine net (actual) counts from source]; record background count on Data Table 11-4-2.

3. Place first source in sample holder, not directly under GM tube, but allowing enough room to put other source in later; take a 1 to 2 minute count - long enough to get at least 1000 counts; record data.
4. Without moving or disturbing first source, put second source in sample holder; obtain count on both sources; record data.
5. Without disturbing source 2, remove source 1 and repeat counting procedure with only source 2 in place; record data.
6. Calculate resolving time (use Equation 11-4-1).

GM COUNTER EFFICIENCY

1. Obtain background count (as described above).
2. Count all 5 sources for 1-2 minutes each on at least 2 shelves (preferably 3) in sample holder. Record data in Data Table 11-4-3; make all indicated calculations. % efficiency = net cpm from source ÷ source dpm x 100 (Equation 11-4-4).
3. For each of the shelves counted, compute geometry factor (use Equation 11-4-5). Explain why G and % efficiency differ for each shelf.

DATA TABLE 11-4-1

GM VOLTAGE	COUNTS	TIME	COUNTS PER MINUTE

1. Plot above data on graph paper.
2. Determine operating voltage: ________ volts.
3. Use Equation 11-4-2; calculate % slope of plateau:
   % slope = ________.
4. Plateau length is ________ volts.
5. Breakpoint voltage is ________ volts.

DATA TABLE 11-4-2

Background = ________ counts in ________ minutes
= ________ cpm_{bkg}.

Count rate for source 1 = R_1 = ________ counts in ________ minutes = ________ cpm - ________ cpm_{bkg} = ________ cpm_{net}.

Count rate for both sources = R_{12} = ________ counts in ________ minutes = ________ cpm - ________ cpm_{bkg} = cpm_{net}.

Count rate for source 2 = R_2 = ________ counts in ________ minutes = ________ cpm - ________ cpm_{bkg} = ________ cpm_{net}.

Calculate resolving time T (use net counts only):

$$T = \frac{R_1 + R_2 - R_{12}}{2 R_1 R_2} = \frac{\underline{\hspace{2cm}} + \underline{\hspace{2cm}} - \underline{\hspace{2cm}}}{2 \times \underline{\hspace{2cm}}}$$

T = ________.

DATA TABLE 11-4-3

Background = ________ counts in ________ minutes

= ________ cpm_{bkg}.

SHELF 1: G = ____________.

	^{14}C	^{90}Sr - ^{90}Y	^{204}Tl	Gamma Source	α Source
Known Strength, dpm					
cpm from Source					
Background cpm					
Net cpm					
% Efficiency					

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SHELF 2: G = ____________.

	^{14}C	^{90}Sr - ^{90}Y	^{204}Tl	Gamma Source	α Source
Known Strength, dpm					
cpm from Source					
Background cpm					
Net cpm					
% Efficiency					

SHELF 3: G = ____________:

	^{14}C	^{90}Sr - ^{90}Y	^{204}Tl	Gamma Source	α Source
Known Strength, dpm					
cpm from Source					
Background cpm					
Net cpm					
% Efficiency					

REFERENCE MATERIALS

Chase, G.D. and Rabinowitz, J.L. Principles of Radioisotope Methodology. Minneapolis, MN: Burgess Publishing Company, 1965.

Glasstone, Samuel, ed. Sourcebook on Atomic Energy. 3rd ed. New York: Van Nostrand-Reinhold, 1967.

International Commission on Radiological Units and Measurements (ICRU). "Radiation Protection Instrumentation and Its Application," Report No. 20. Washington, DC: National Bureau of Standards, 1971.

Maushart, R. Radiation Protection Instrumentation. Oxford: Pergamon Press, 1972.

Quimby, E.H. and Feitelberg, S. Radioactive Isotopes in Medicine and Biology, Vol. 1. New York: Lea and Febiger, 1965.

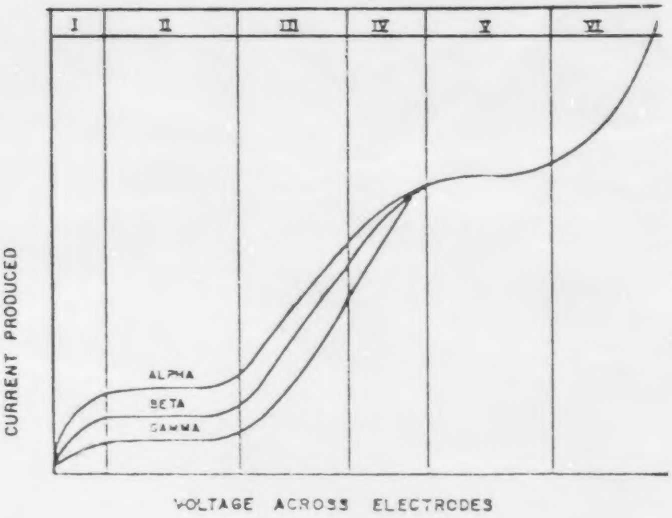
TEST

NOTE TO INSTRUCTOR:

Students/trainees need a slide rule, calculator and/or square root charts for this test.

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space. (Note: a slide rule, calculator, or square root chart is needed.)

1. Label the regions that are numbered I-VI on the graph below.



2. In Region V, what relationship, if any, exists between pulse height and the number of ion pairs producing the pulse? Why is the pulse size so large in Region V? ______________________________

____

____

____

____

____

3. Why is Region IV not used for radiation detection and measurement?

____

____

____

4. Draw and label a typical curve for a detector operating in the Geiger-Mueller region. Label starting potential, threshold, plateau, breakdown voltage and continuous discharge.

5. Why can't a G-M detector be used as a dose or exposure rate meter?

____

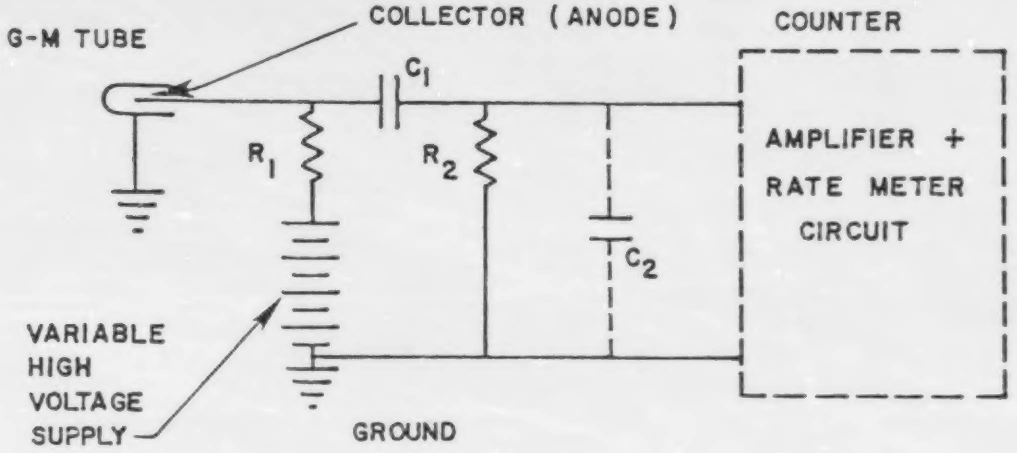
____

6. How could you tell that breakdown voltage had been reached?

____

____

7. A schematic diagram of a Geiger-Muiller counting circuit is shown below. Use the diagram to respond to Parts a-d of this question.


G-M TUBE
COLLECTOR (ANODE)
COUNTER
C_1
R_1
R_2
C_2
AMPLIFIER +
RATE METER
CIRCUIT
VARIABLE
HIGH
VOLTAGE
SUPPLY
GROUND

a. What is the purpose of C_1?

b. What is the purpose of R_1?

c. How are ions collected?

d. What happens when positive ions reach the cathode of the tube?

8. What causes multiple discharges in the G-M counter?

9. Describe the process of external quenching? What limitation does this process place on the detector? ____________________

10. Briefly describe internal quenching. ____________________

11. What is dead time? ____________________

12. What is recovery time? What is happening in the G-M circuit during recovery time? ____________________

13. List three factors that influence G-M detector counting efficiency.

a. ____________________

b. ____________________

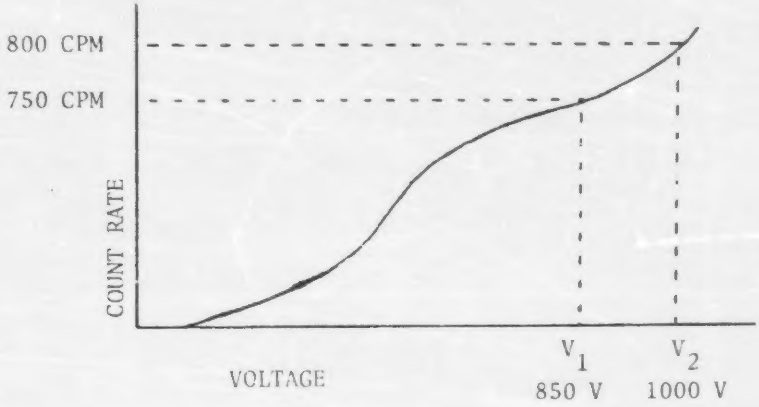
c. ____________________

14. List two limitations of the G-M detector.

a. ____

b. ____

15. Use the graph below to calculate % slope.



16. Calculate efficiency (% EFF) of the counting system given the following: A one liter sample of reactor coolant water has been shown to contain ^{134}Cs with an activity of 3.0 μCi; and a net count rate of 1575 cpm.

17. Calculate the geometry factor given the following: A small radiation located in a standard sample holder 4 inches from the window of a G-M tube; tube radius is .5 inches.

18. Calculate resolving time, given the following information: In the laboratory, two radiation sources were first counted separately and then simultaniously with these results --

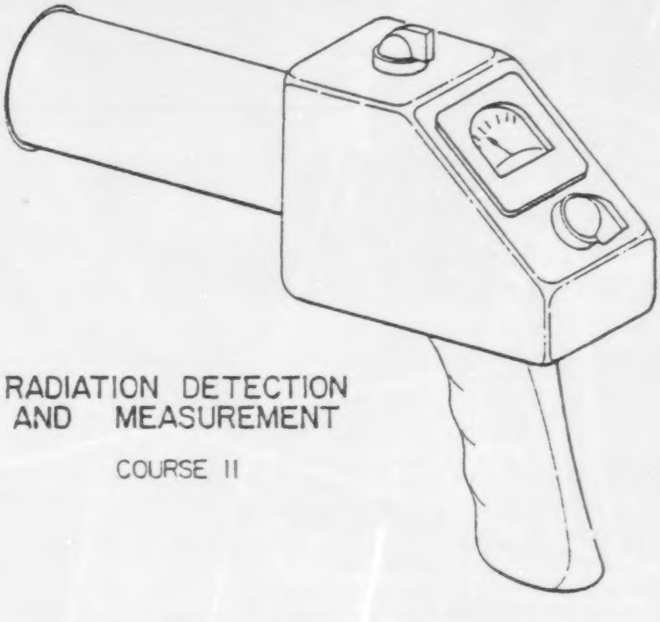
R_1 (first source counted by itself) = 3000 cpm

R_2 (second source counted by itself) = 3400 cpm

R_{12} (both sources counted together) = 6130 cpm

NUCLEAR TECHNOLOGY

MODULE 11-5 SCINTILLATORS, SEMICONDUCTORS AND SPECTROSCOPY.



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
⇨ MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

In the previous three modules of this course, gas-filled detectors and their applications to radiation measurements were presented. These gas-filled detectors are widely used, not only in laboratory counting of radioactivity, but especially as portable radiation monitors. But they constitute only one class of radiation detectors.

A second class, which is the subject of this module, is the solid-state detectors. These also find wide application in laboratories for radiation energy determination (i.e., spectroscopy) and/or pulse-height analysis - methods for specifically identifying radionuclides by their energy of radiation.

In Modules 11-2, "Ionization Detectors," and 11-3, "Proportional Counters," ionization chambers and proportional counters were shown to be useful for differentiating between alpha, beta and gamma radiation on the basis of the amount of ionization, or pulse size, produced by incident radiation. Theoretically, given the proper and suitable electronics, any one of these radiations can be sorted by energy, since pulse size is proportional to deposited energy in the chamber. For alpha particles, this theory is satisfactory, since only a very small chamber is needed to collect all the energy associated with an alpha particle. For beta particles, the problem becomes more difficult because a beta particle may have a very long range in gases - perhaps as great as five or more meters. Thus, a very large chamber would be needed to collect all the energy associated with a beta particle and the theory becomes obviously impractical. The problem is even more pronounced with gamma rays because of the enormous distance they may travel in gas before interacting.

Solid-state detectors, with their higher density, can be used for spectroscopy because of a much larger probability that gamma rays will interact within a fairly thin or small detector. On the basis of density alone, there is a several thousand-fold increase in the probability of interaction. In addition, solid-state detectors have higher atomic numbers than gases and, as shown in Course 1, "Radiation Physics," the probability of interaction is greater.

This module covers only a couple of the solid-state detectors: scintillation and semiconductor devices, used as pulse counters and mean-level devices. (Others, such as photographic films, TLD's and other integrating devices are discussed in Module 11-6, "External Personnel Dosimeters for Beta-Photon Monitoring.")

MODULE PREREQUISITES

The student should have a basic understanding of atomic and nuclear physics and radiation interactions with matter as might be obtained from satisfactory completion of Course 1, "Radiation Physics," in this series. In addition, the student should have a basic understanding of physics and electronics as might be obtained from a technical physics course. The student must also have satisfactorily completed Modules 11-1 through 11-4 in this series ("Statistics of Radiation Counting," "Ionization Detectors," "Proportional Counters," and "Geiger-Mueller Counters").

OBJECTIVES

Upon completion of this module, the student should be able to:

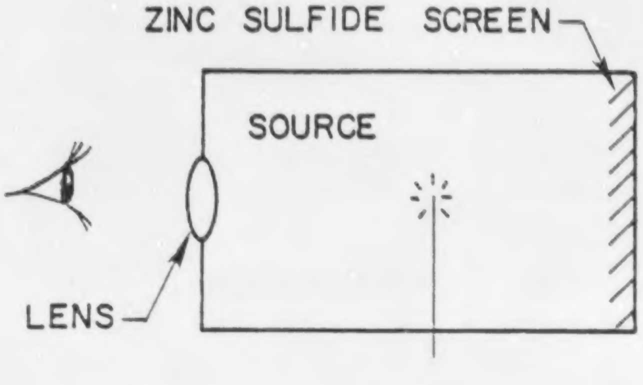
1. List the four basic types of scintillation counters and briefly describe the characteristics of each.
2. Describe how scintillators are used for detecting neutrons.
3. Describe how a photomultiplier tube operates; calculate gain, given a) number of stages and b) secondary emission ratio.
4. Draw block diagram of scintillation counting system and gamma spectroscopy system.
5. Describe three basic categories of semiconductor detectors; discuss their applications to radiation detection and measurement.
6. Briefly describe how gamma ray spectroscopy is performed; carry out a gamma ray spectroscopy procedure (use both a scintillator and a GeLi detector, if available).
7. Define the following terms:

   Scintillations.
   Photomultiplier tube.
   Luminescence.
   Phosphorescence.
   Activator.
   Photocathode.
   Dynodes.
   Secondary emission ratio.
   Well crystal.
   Semiconductor.
   p-n junction.
   Depletion layer.
   GeLi detector.
   Pulse-height analysis (PHA).

SCINTILLATION DETECTORS

Scintillation detectors are among the oldest but still most widely used radiation detectors in nuclear science. Only three years after his discovery of radioactivity, Henri Becquerel observed and carefully studied the emission of light from various crystalline substances exposed to radiations from abrasions, including zinc sulfide and diamonds. These flashes of light were named scintillations, this word being coined from the Latin word "scintilla" which means spark. Becquerel's studies were conducted in 1899, and his results formed the basis for studies by others, including the Curies, Sir William Crookes and Ernest Rutherford. By 1903, Crookes had devised an instrument he called a "spinthariscope" (Figure 11-5-1), that could be used to visualize individual scintillations, or flashes of light, from alpha particles.

Figure 11-5-1.
Schematic drawing of a spinthariscope.



By 1908, German physicist E. Regener was using scintillations in diamonds to detect alpha particles, counting each flash that he observed by eye. In 1944, the photomultiplier tube was invented which made possible amplification and electronic counting of scintillations.

Scintillation detectors make use of a physical property known as luminescence, which is absorption of energy (in this case, from nuclear radiations) by a substance followed by re-emission of that energy in the form of visible light. In most cases, this light is emitted virtually instantaneously after absorption of the radiation, but in some cases it may be delayed. If emission of light occurs within 10^{-8} seconds (0.01 μsec) of the time of excitation, the process is called luminescence. If emission of light lasts longer than or takes place after 10^{-8} seconds, the process is called phosphorescence.

Unfortunately, the terms luminescence and phosphorescence are often improperly used (or used in slightly different ways by different scientists). Hence, the term "phosphor" is commonly used in a generic way to refer to any "scintillator," and these two are often used interchangeably. Actually, phosphorescent materials are seldom used or are of limited use in nuclear counting work because of their long resolving times.

Generally, scintillators can be divided into four categories:

Inorganic crystals.
Organic crystals.
Liquid scintillators.
Plastic scintillators.

INORGANIC CRYSTALS

Inorganic crystals are inorganic salts, primarily alkali halides, that form regular crystalline lattices. If the lattice has certain imperfections, such as physical defects or deliberately introduced impurities, these will cause the crystal to scintillate, or emit light, when exposed to radiation.

Inorganic crystals, the most commonly used of all scintillators, are in widespread use as gamma detectors. The most common inorganic crystal used for scintillation counting is sodium iodide. It is activated by deliberate inclusion of a small amount of thallium as an activator (a trace quantity of a substance that imparts luminescence to crystals).

Other important inorganic crystals used in scintillation counting are described in Table 11-5-1, along with some of their more important characteristics.

TABLE 11-5-1

PROPERTIES OF COMMON SCINTILLATORS
(Activators in Parentheses)

Material	Type	Main Use	Density g/cm³	Wave Length of Light Emitted, Nanometers	Duration of Light Emission, usec	Relative Pulse Height
NaI(Tl)	Inorganic Crystal	Gamma Counting	3.7	410	0.25	2.1
KI(Tl)	"	"	3.2	400	5.0	0.8
CsI(Tl)	"	"	4.5	White	1.0	0.5
ZnS(Ag)	"	Alpha Counting	4.1	450	10.0	2.0
LiI(Eu)	"	Neutron Detection	4.1	440	1.0	0.75
Anthracene	Organic Crystal	Beta Counting	1.3	440	0.03	1.0
Stilbene	"	"	1.2	410	0.008	0.6
Plastic	Solid	"	1.1	350-450	0.005	0.35
p-Terphenyl	Liquid	"	0.9	350-450	0.003	0.35

By looking at the first five scintillators in Table 11-5-1, the advantages of NaI(Tl) become apparent. Note that all are phosphorescent, that is, all have light decay times (duration of light emission) greater than 10^{-8} seconds. The NaI(Tl) has the shortest decay time and, for a given type of radiation, it also produces the largest relative pulse height - the greatest amount of light output. It is also fairly dense, with good stopping power for gamma rays. In addition, NaI(Tl) can be made in the form of very large crystals, in some cases as great as 50 cm (20"). Finally, it is highly transparent, providing good light transmission properties with a suitable refraction index.

Inorganic crystals, such as NaI(Tl), are made by carefully growing crystals in a controlled environment to produce uniform, regular, crystalline structures with few imperfections. During the crystal growing process, minute controlled quantities of activator are introduced. After the crystals have been grown, they are dried, cleaved to an appropriate size, and polished. Then, in the case of most inorganic crystals, they are encased in light-tight cans.

These cans are usually made of thin, low Z metals, such as beryllium or aluminum, to minimize absorption of x and gamma rays. In the case of NaI(Tl), which is hydroscopic, the canning procedure serves to seal out moisture as well. Also, a reflector provided around the crystal improves its efficiency by bouncing light back into it.

Visual examination and manual counting of scintillations in a crystal is a slow, tedious, inexact and very limited method of radiation detection. Hence, scintillators are used in conjunction with an electronic device known as a photomultiplier (Figure 11-5-2), which is usually canned in with the scintillator.

A photomultiplier tube is a combination of 1) an ordinary photoelectric tube (or cell) with 2) an electron multiplying device.

Light from a scintillator is transmitted or directed to the face (photocathode) of a photomultiplier tube. This photocathode is made of a metal or metallic compound that emits electrons when struck by light (as a light meter used in photography does). Electrons emitted by the photocathode are attracted to the first of a series of electrodes, whose primary function is secondary emission of electrons.

These secondary emissions, called dynodes, are maintained at a positive potential with respect to the photocathode. Each dynode (there is a whole series, usually about ten) emits more electrons when struck by an original electron, and acts as an electron multiplier. A gradient of approximately 100 volts is maintained between each successive pair of dynodes, and the dynodes are shaped so that electrons emitted from one are focused onto the text.

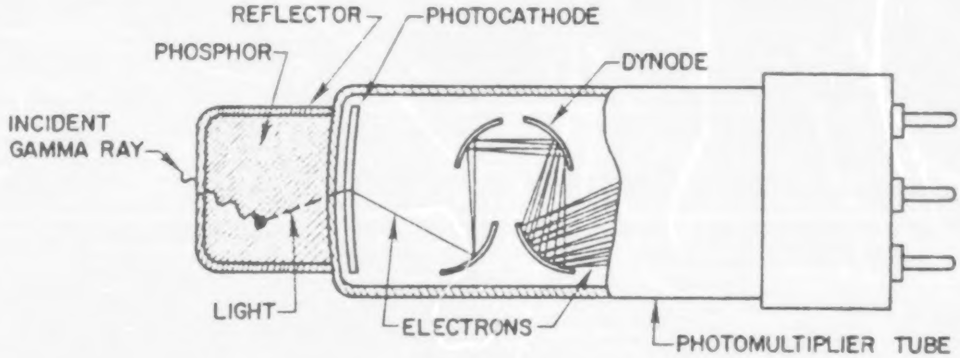


Figure 11-5-2. Scintillation phosphor and photomultiplier tube.

The typical multiplication, or gain, of a photomultiplier tube is about 10^6. Thus, electrons emitted by a photocathode are amplified about a million times, then collected at the anode. The number of electrons collected is directly proportional to the number of electrons emitted by the photocathode. And the number emitted by the photocathode is directly proportional to the amount of light that fell on it.

Thus, the photomultiplier serves two main functions: 1) It converts light from the scintillator into electrons - current. 2) It amplifies the current it produces.

The amount of amplification is a function of many factors, including number of dynodes and multiplication between dynodes. The ratio of electrons emitted from a given dynode to the number emitted from the previous dynode is known as the secondary emission ratio, which ranges from about 2 to 7, depending on the materials and construction at the dynode and on the energy of the electrons. The total gain and multiplication of a photomultiplier tube can be calculated as follows:

$$M = \delta_1 \delta_2 \cdot \cdot \cdot \delta_n$$

Equation 11-5-1
(Equation = Eq.)

where

M = total gain or multiplication.

δ = secondary emission ratio for a particular pair of dynodes.

n = number of stages, or dynodes.

Since, in any given photomultiplier tube, the secondary emission ratio is constant between adjoining pairs of dynodes,

Equation 11-5-1 simplifies to:

$$M = \delta n \qquad \text{Eq. 11-5-2}$$

or, for a gain of 10^6 in a 10-stage photomultiplier, $\delta = 4$. Note: The student should verify this number.

The design and construction of photomultipliers is highly complex and in many respects more of an art than a science. Photocathodes are commonly made of alternate layers of antimony and cesium. Operating voltages for photomultipliers range from about 1000 to 1500, with some tubes requiring in excess of 2000 volts.

Although sodium iodide is the most commonly used scintillator, the other scintillators shown in Table 11-5-1 are also used. ZnS(Ag) is made in the form of small crystals and is particularly suited to alpha particle detection. Since it is not hygroscopic, it does not need to be hermetically sealed, and it can be applied in the form of a paint or similar material brushed onto a surface. ZnS(Ag) is found in portable alpha survey meters, with the scintillator being applied to the back side of a thin window, or sometimes painted right onto the face of the photomultiplier itself.

In a commercial alpha probe that uses ZnS(Ag) as the detector behind a 1 mg/cm² aluminized mylar window (Figure 11-5-3), the window is covered with a metal mesh to protect it from damage. In the unit at left (in the figure), light from the phosphor is scattered into the detector; in the unit at right, light is transmitted through a "light pipe" into the photomultiplier.

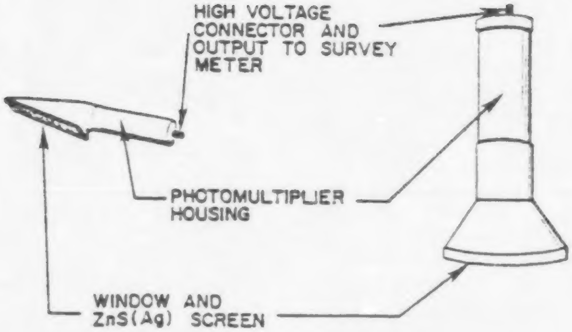

HIGH VOLTAGE CONNECTOR AND OUTPUT TO SURVEY METER
PHOTOMULTIPLIER HOUSING
WINDOW AND ZnS(Ag) SCREEN

Figure 11-5-3. Commercially available alpha probes for use with portable survey meters.

LiI(Eu) is a scintillation material employed in neutron detection and counting. The detector, which may be enriched in ^{6}Li, takes advantage of alpha particles produced inside the scintillator by ^{6}Li (n,α) ^{3}H reaction.

The LiI(Eu) crystal is relatively insensitive to gamma radiation, and can be removed electronically because of its smaller pulse. Hence, the LiI(Eu) detector is sometimes used instead of a BF_3 proportional counter. The photomultiplier tube (Figure 11-5-4) is housed outside the polyethylene moderator, and light from the scintillator is passed through an acrylic plastic light pipe to it. The unit weighs about 7.5 kg (18 lbs).

ORGANIC CRYSTALS

Organic crystals comprise a second category of scintillators and, because of their low Z number and relatively low density (Table 11-5-1), are used primarily for beta counting.

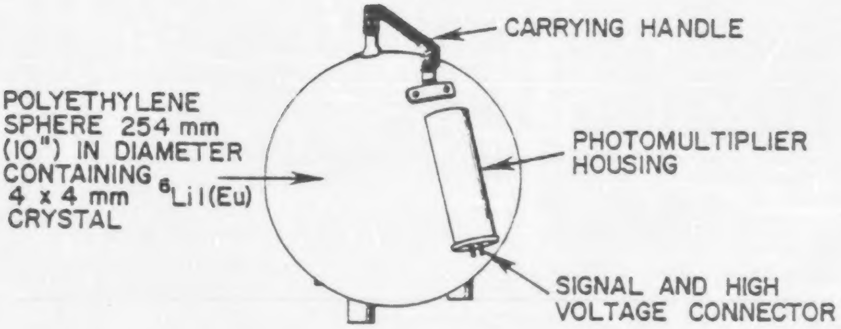

CARRYING HANDLE
POLYETHYLENE SPHERE 254 mm (10") IN DIAMETER CONTAINING 4 x 4 mm ⁶LiI(Eu) CRYSTAL
PHOTOMULTIPLIER HOUSING
SIGNAL AND HIGH VOLTAGE CONNECTOR

Figure 11-5-4. Neutron detector using ^{6}LiI(Eu) crystal for use with portable survey meters.

Organic chemicals contain carbon, and most carbon compounds are non-ionic in nature. Carbon combines most readily with hydrogen and oxygen, hence most organic compounds have low average Z.

Organic crystal scintillators are generally aromatic hydrocarbons containing benzene rings. Perhaps the most common organic crystal is anthracene, a coal tar derivative with the empirical formula $C_{14}H_{10}$ and the structural formula shown in Figure 11-5-5. Stilbene, another fairly common organic crystal scintillator has the formula $C_{14}H_{14}$ and is more correctly known as symmetrical diphenylethylene. Both of these organic crystals have six carbon benzene (phenyl) rings in their structure.

Unlike the inorganic crystals, which emit light when an electron drops from a higher energy state to a lower energy state, the luminescence process in organic crystal scintillators occurs on a molecular level. Radiation raises the energy level of the molecule from the electronic ground state to the excited state, with the whole molecule ultimately returning to the ground state by emitting its excess energy in the form of light. The organic crystal is transparent to the released light.

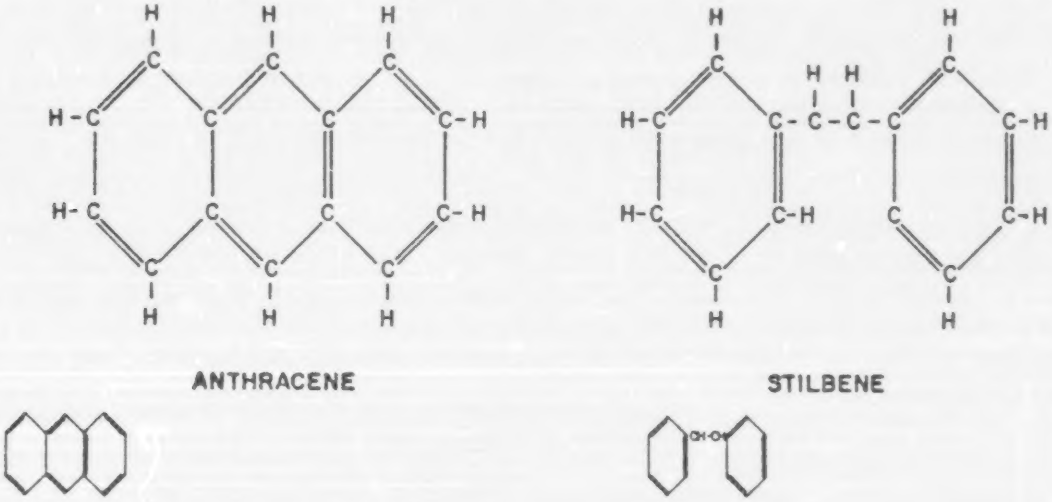


Figure 11-5-5. Structural formulae of anthracene and stilbene. The outline formula is shown to the lower left of the complete drawing.

LIQUID SCINTILLATORS

Scintillation crystals discussed previously have a very limited usefulness for the detection of weak beta particles, because the particles are absorbed by self-absorption or by the surroundings before they can interact with the scintillator. With liquid scintillators, the source can be placed directly in the scintillator, thereby increasing sensitivity.

Liquid scintillators are generally a solution of an organic scintillator material in an organic solvent. The active portion of these is an aromatic hydrocarbon similar to those used for organic crystal scintillators. Hence, the mechanism of interaction with radiation to produce light is the same as for organic crystals.

The great advantage of liquid scintillators is that they can be mixed with water or other solutions and used to count low-energy beta particles from the decay of ^{14}C (0.156 MeV_{max}) and ^{3}H (0.0186 MeV_{max}). Counting such low-energy radiations would be virtually impossible otherwise. For example, the maximum range of a tritium beta particle in water is only about 0.025mm (0.001 inches). Thus, unless tritium is in intimate contact with the scintillator, no interaction occurs.

There are many liquid scintillation mediums, but most commonly used is 5 grams of p-terphenx dissolved in one liter of solvent (dioxane, in most cases). Other mediums have very formidible scientific names and are better known, therefore, by their trade names, or initials. Two of these common liquid scintillation phosphors are PPO (chemical name: 2,5-diphenyl-oxagole) and BBOT [chemical name: 2, 5-bis2(5-t-bulyl-benzoxagolyl)-thiophene], which are usually dissolved in xylene, dioxane or other aromatic solvents. Note that both PPO and BBOT, as well as p-terphenyl, are compounds containing the six carbon benzene (phenyl) rings.

PLASTIC SCINTILLATORS

This final category of scintillators, which is seldom used, is essentially a solid organic solution. Plastic scintillators, sometimes used for beta counting, have much lower efficiency than organic crystals of comparable size. They can be made much larger, however, and are more easily handled and shaped than organic crystal scintillators.

SCINTILLATOR CONSIDERATIONS

An important consideration, with regard to all scintillators, is efficient transfer of light (produced in the scintillator) to the photocathode of the photomultiplier tube. Most scintillators used in laboratory counting are in the shape of a solid right cylinder, or a right cylinder with a hole cut in it (Figure 11-5-6), ergo, a well crystal.

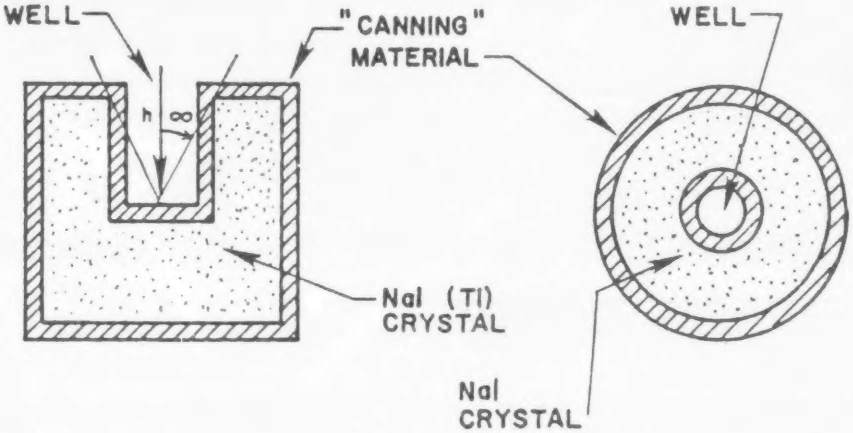


Figure 11-5-6. Schematic drawing of NaI(Tl) well crystal. Note that a liquid source can be counted in a test tube placed inside well.

Well crystals are designed so that a sample can be inserted inside detectors, achieving a counting geometry approaching 4π. The right cylinder shape of most scintillators has lead to their specification of size in terms of diameter and height. Hence, a typical NaI(Tl) crystal, three inches in diameter and three inches high, would be known as "3 by 3" or "3x3." Other common dimensions for NaI(Tl) crystals are 1x1, 1x2, 2x2, 4x4 and 5x4.

To obtain maximum efficiency, the crystal is optically connected to the photomultiplier tube to minimize light losses. Also the crystal itself is surrounded by a reflector to reflect the light that would otherwise escape back into the crystal (Figure 11-5-2). The face of the scintillator is directly connected to the photocathode window by an optical coupling agent or cement, and the photocathode window is as transparent to light as possible in the region of wavelengths emitted by the scintillator. If the scintillator is to be directly connected or coupled to the photomultiplier, the two should be the same size to minimize losses.

If direct coupling is not possible (for example, because of the geometry of the detector), or if it is desirable to spread the light over a larger area or concentrate it to a smaller area (as would be the case when the scintillator and photomultiplier are of different sizes), a "light pipe" may be used (Figure 11-5-3). Light pipes are merely materials that pass light - usually acrylic plastics or, in some instances, fiber optics. The latter is particularly useful where the light must be bent or moved in an irregular path.

SCINTILLATOR APPLICATIONS

Table 11-5-1 shows some of the applications of scintillators and, as can be seen, this type of detector is quite versatile. Large NaI(Tl) detectors (Figure 11-5-7) can be used for low-level counting of gamma rays. They will detect even high-energy photons with a fairly high degree of efficiency.

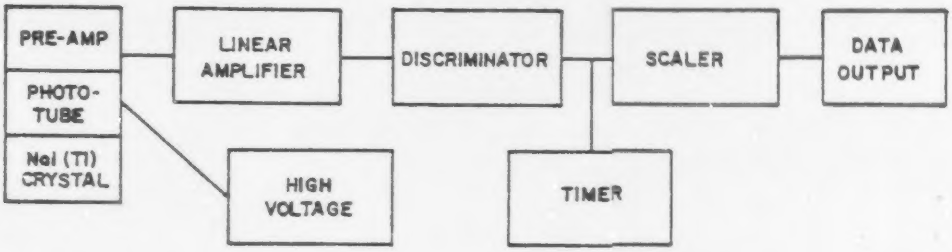


Figure 11-5-7. Sodium iodide scintillation counting system.

Liquid scintillation counting, however, is more complex than solid scintillation counting. Thermal emission of electrons from the photocathode, a normal phenomenon, will cause spurious counts and prevent low-level counting of ^{14}C or ^{3}H in liquid samples. Some relief can be obtained by cooling the photomultiplier tube, but electronic methods are often used as well.

In typical liquid scintillation counting (Figure 11-5-8), the scintillations produced by the betas are seen by two phototubes, and simultaneously appear at the inputs to the monitor and analyzer amplifiers. If sufficiently large, pulses from the monitor pass discriminator C and go into the coincidence logic circuit. Pulses from the analyzer amplifier, which exceed the setting of discriminator A but not B, pass through the logic and are counted. Pulses that exceed the level of B are killed in logic. Only pulses that occur simultaneously in both photomultiplier tubes are counted. (Very large pulses - as from cosmic rays - are not counted because they have been killed in logic; thermal noise pulses are not counted because they do not ordinarily occur simultaneously in both photomultipliers.)

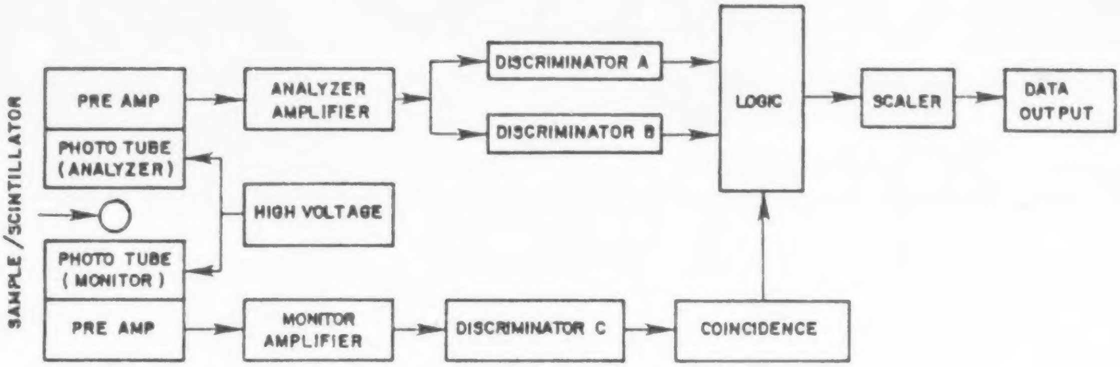

SAMPLE /SCINTILLATOR
PRE AMP
PHOTO TUBE (ANALYZER)
PHOTO TUBE (MONITOR)
PRE AMP
ANALYZER AMPLIFIER
HIGH VOLTAGE
MONITOR AMPLIFIER
DISCRIMINATOR A
DISCRIMINATOR B
DISCRIMINATOR C
LOGIC
COINCIDENCE
SCALER
DATA OUTPUT

Figure 11-5-8. Simplified block diagram of a single channel liquid scintillation counting system using coincidence logic.

SEMICONDUCTOR DETECTORS

In the early 1960's, the entire radiation detection and measurement field was revolutionized by the development of semiconductor detectors. These devices have been, thus far, mainly applied in the field of particle and gamma spectroscopy, although they can also be used as pulse counters. There are three major categories of semiconductor detectors:

Diffused p-n junction.
Surface barrier.
Lithium drifted.

DIFFUSED p-n JUNCTION DETECTOR

The diffused p-n junction detector derives its name from both its method of manufacture and from its semiconductor properties. A semiconductor is a solid crystalline material with electrical conductivity somewhere between that of good conductors, such as metals, and insulators. When exposed to radiation, the electrical conductivity of a semiconductor changes, and these changes can be measured and correlated with the radiation exposure. In a diffused p-n junction detector, a wafer or slice of p-type semiconductor (usually silicon or germanium) has a layer of n-type material, such as phosphorous, deposited on its surface. The semiconductor crystal is heated and phosphorous diffuses into it to form a p-n junction just below the surface.

The letters "p" and "n" refer to the electron configuration of the material and result from the addition of appropriate impurities during manufacture of the semiconductor crystal. An n-type material has an excess of electrons in the crystalline structure, hence "n" for negative. A p-type material has an excess of "holes" in the crystalline structure - places where an electron ought to be but, instead, a net positive charge exists because there are insufficient numbers of electrons. In this case, "p" stands for positive.

N-type materials are made by adding an impurity element into the semiconductor with a larger number of valence electrons than the semiconductor material has. For example, silicon has four valence (or outer shell) electrons. Introduction of arsenic, with five valence electrons, provides one extra. This extra electron exists as a free electron in a higher energy state than the normal valence band electrons, but not sufficiently high in energy to move freely throughout the crystal (in the conduction band) and, therefore, make it conductive.

A p-type material is made by incorporating lower valence impurities, for example aluminum or boron (with a valence of three), into the crystal of a four-valent element such as silicon.

The natural action of the combined materials in a diffused p-n junction detector (Figure 11-5-9) tends to align the excess electrons on one side of the junction and the holes on the other, thus producing a buildup of potential across the junction. This small potential difference can be enhanced by applying an additional external voltage to the crystal to oppose the natural movement of electrons and holes - the so-called "reverse bias" - which is applied by connecting the positive pole of the battery to the n side, and negative pole to the p side. This setup produces a depletion layer containing electrons or holes, which is the sensitive volume of the detector, and is analagous to the gas in a gas-filled detector. The depth of the depletion region, or layer, is a function of the reverse-bias voltage.

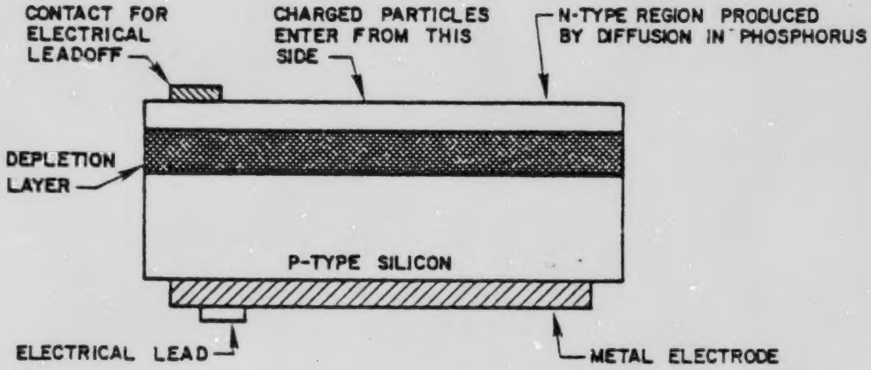

CONTACT FOR ELECTRICAL LEADOFF
CHARGED PARTICLES ENTER FROM THIS SIDE
N-TYPE REGION PRODUCED BY DIFFUSION IN PHOSPHORUS
DEPLETION LAYER
P-TYPE SILICON
ELECTRICAL LEAD
METAL ELECTRODE

Figure 11-5-9. Diffused p-n junction detector.

Charged particles entering the depletion layer produce electron-hole pairs (analagous to ion pairs in gases), which move quickly towards the positive or negative electrodes, thus producing a current pulse that can be amplified and electronically measured. As with ionization chambers, proportional counters and scintillators, the size of the pulse is proportional to the energy of the particle.

The diffused p-n junction semiconductor detector is most useful for detecting charged particles, since the depletion layer is at most a few millimeters deep. Other than spectroscopy applications, which is discussed later, these semiconductors have relatively few laboratory counting or other applications as pulse counters. Because of their low efficiency and high cost, other types of detectors are simply superior for straight pulse counting applications.

SURFACE-BARRIER DETECTOR

The surface-barrier detector constitutes a second category of semiconductor detectors. Its principle of operation is derived from the fact that at the surface of a pure crystal of silicon or germanium, an electric field exists and serves to exclude both holes and electrons from a thin region near the surface. For n-type crystals, then, this field would repel electrons from the region. If a metal is joined to the surface of the crystal, the free electrons are repelled, which results in a concentration of holes directly beneath the surface (Figure 11-5-10).

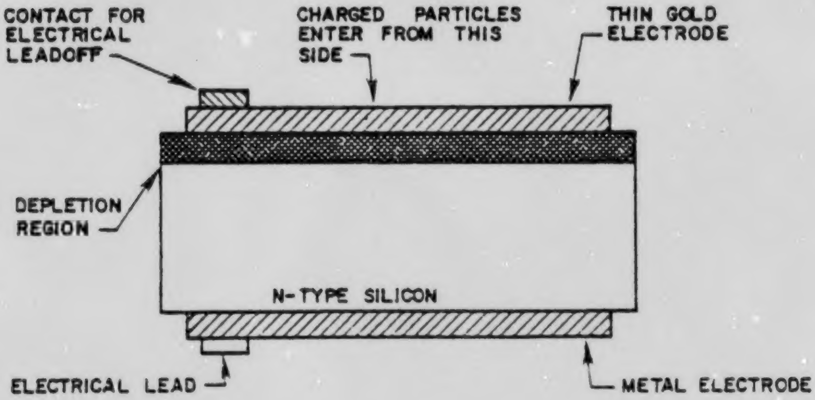

CONTACT FOR ELECTRICAL LEADOFF
CHARGED PARTICLES ENTER FROM THIS SIDE
THIN GOLD ELECTRODE
DEPLETION REGION
N-TYPE SILICON
ELECTRICAL LEAD
METAL ELECTRODE

Figure 11-5-10. Surface-barrier detector.

When a reverse bias is applied, a depletion layer is produced. Ordinarily, gold about $2x10^{-8}$ cm thick (40 mg/cm^2) is used as the surface metal for n-type silicon and provides the negatively biased electrical contact. A metal, such as aluminum, is used for the positively biased contact.

Surface-barrier detectors give better energy resolution than diffused p-n junctions, but the depletion region is not as thick. Hence, the surface-barrier detector is useful for lower energies; that is, it does not have as wide an energy detection range as a junction-type detector.

Both surface-barrier and junction-type detectors show significant advantages over gas-filled detectors and scintillators. Background from cosmic rays or other external sources of gamma rays are extremely low because of the small size of the detector. This characteristic enhances their use for low-level particle counting. Resolution times are extremely short, generally on the order of nanoseconds. And, most importantly, the ability to accurately differentiate energy deposited by particles - resolution - is superb, as will be discussed in the section on spectroscopy.

There are also disadvantages to semiconductor detectors. They are fragile and require careful handling and storage. Output signals are small and require electronic amplification. Electronic noise, either from the amplifier or inherent in the detector, may be a limiting feature. And, should a detector become contaminated, as may occasionally happen, decontamination is impossible because of their construction and fragile nature.

LITHIUM-DRIFTED DETECTOR

The final category of semiconductor detectors is the lithium ion drifted junction detector, introduced by the American physicist E.M. Pell in 1960. This detector is made by applying lithium (Li), which acts as an electron donor, to the surface of a low-resistivity p-type material such as germanium or silicon. The crystal is heated, allowing the Li to diffuse into the crystal. This action forms an n-type layer. The crystal is heated again, this time to about 150°C, with a reverse bias of several hundred volts applied. The electric field causes the lithium ion to drift into the crystal, forming a wide layer in which the Li electron donor exactly compensates the p-type material. Thus, an effective intrinsic semiconductor layer is formed that is relatively thick (Figure 11-5-11); indeed, thicknesses of more than 1 cm can be achieved. Lithium-drifted detectors are useful for gamma spectroscopy.

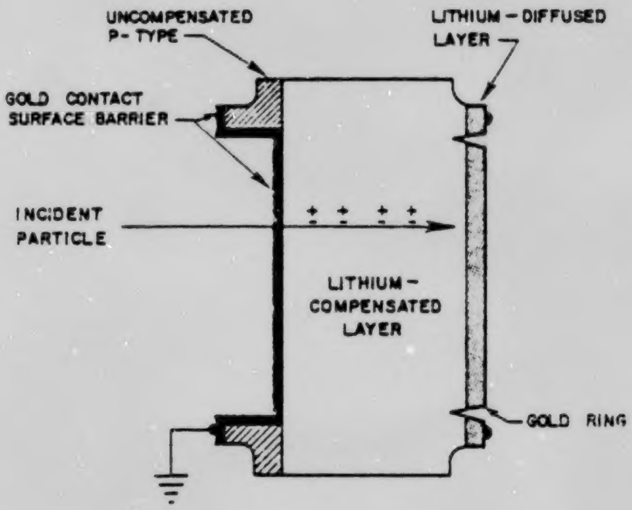

UNCOMPENSATED P-TYPE
LITHIUM-DIFFUSED LAYER
GOLD CONTACT SURFACE BARRIER
INCIDENT PARTICLE
+ + + +
LITHIUM-COMPENSATED LAYER
GOLD RING

Figure 11-5-11.
Lithium ion drifted detector.

Lithium-drifted silicon detectors, designated as SiLi (pronounced "silly"), can be operated at room temperature; however, they have relatively low efficiencies because of the low Z and hence low stopping power of their major component silicon (Z=14). On the other hand, a GeLi (pronounced "jelly") detector, consisting mostly of germanium (Z-28) has a much greater stopping power and hence much greater efficiency.

To reduce background noise, GeLi detectors must be operated at very low temperatures and must be kept cold with liquid nitrogen; these detectors are appropriately designed to hold this coolant. Many GeLi detectors have been ruined by allowing them to warm up to room temperature while under voltage.

The limitations and advantages of diffused junction detectors, discussed above, generally apply to lithium-drifted detectors. In addition, all semiconductors are susceptible to surface degradation, which tends to limit their lifetimes.

SPECTROSCOPY

Every radionuclide emits one or more particles or photons with a unique combination of energies and abundance. If the radiations emitted by an unknown nuclide can be determined, the unknown can be identified and quantified. For example, ^{60}Co emits a beta particle with a maximum energy of 0.314 MeV and two gamma rays, one with an energy of 1.173 and another with an energy of 1.332 MeV. Only ^{60}Co emits this particular pattern of radiation.

Many radionuclides can be identified by their own particular pattern of gamma rays. Iodine-131, for example, emits the following pattern:

Energy (MeV)	% Abundance
0.080	2.6
0.284	5.4
0.364	82.0
0.637	6.8
0.723	1.6

and can readily be identified by it.

Spectroscopy is the process whereby particles or photons are sorted by their respective energies and a plot made of the photons or particles in each energy group (Figure 11-5-12). Only ^{131}I has this particular spectrum; this is its signature, giving positive and absolute identification.

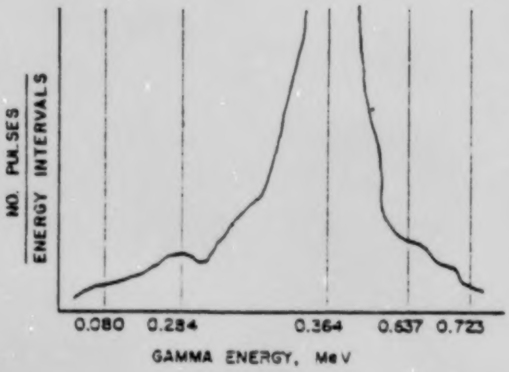


Figure 11-5-12.
Gamma spectrum of ^{131}I as might be obtained with NaI (Tl) crystal. Energy of each peak is identified.

Spectrometry can be used for alphas, betas or gammas, given the appropriate detector. The detector must be one that produces an electronic pulse whose height is proportional to the energy of the radiation that caused the pulse.

Thus, proportional counters, ionization chambers, scintillators and semiconductors could all be used. But Geiger-Mueller counters could not be used because pulse sizes are independent of radiation energy.

In general, scintillators and semiconductor devices are the most commonly used detectors; and gamma spectrometry is done more than alpha and beta spectrometry.

The signal (Figure 11-5-13) from the detector is fed through an amplifier and then into a pulse-height analyzer. There the signal is sorted by size and assigned to one of many available channels or accumulation units. Consider, for example, a series of voltage pulses of various sizes.

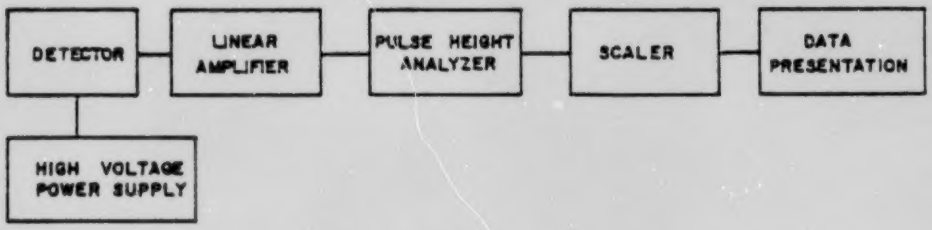


Figure 11-5-13. Spectroscopy system.

One could set the low-level discriminator (LLD) to reject all pulses with a height less than 3 (Figure 11-5-14) and the upper level discriminator to reject all pulses with a height less than 3 (Figure 11-5-14) and the upper level discriminator to reject all pulses with a height greater than 4. Thus, only those pulses with a height between 3 and 4 would be counted. This exemplifies, in its simplest form, a "single-channel" analyzer.

A "multichannel" analyzer will have many - several hundred or even several thousand - such single-channel analyzers,

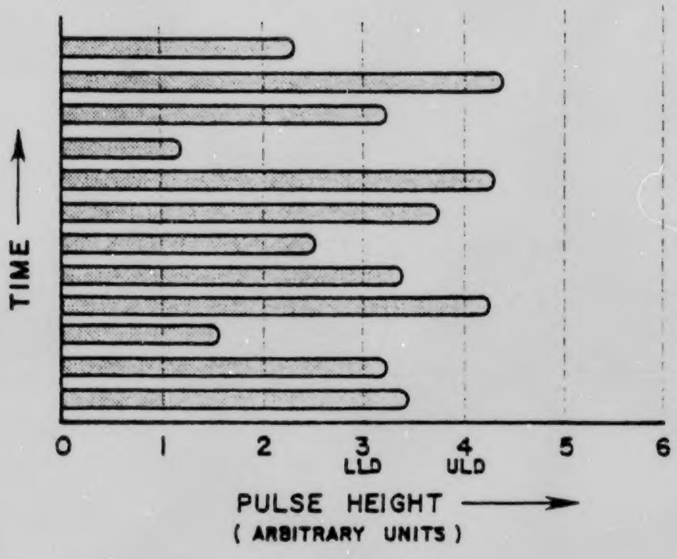


Figure 11-5-14. Pulse height, and the effect of the upper and lower level discrimination. Only those pulses falling between ULD and LLD are counted.

analyzers, each allowing the passage of a pulse with height in a specific range. Each channel or analyzer feeds a separate scaler, and each scaler accumulates counts for a specific pulse-height range.

Since pulse height is directly proportional to energy of the gamma ray or particle causing the pulse, the pulse height can be calibrated in units of energy. A typical expression of pulse-height analyzer calibration, or output, is in units of energy (keV) per channel. A 400-channel analyzer, covering the range 0 to 2 MeV (0 to 2000 keV), would sort pulses into channels in units of $\frac{2000 \text{ keV}}{400 \text{ channels}}$ = 5 keV/channel.

Because of the method of sorting pulses by size, spectrometry or spectroscopy is also known as pulse height analysis (PHA).

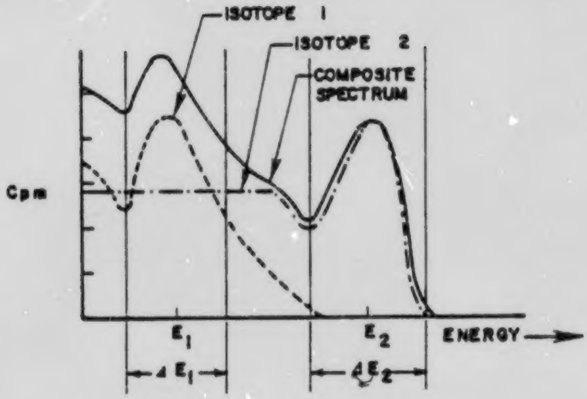
Signals from the pulse-height analyzer drive scalers; data accumulated in scalers can be presented in many ways; e.g., directly on an oscilloscope screen, showing a plot of number of counts per channel verses channel number (effectively, the energy spectrum). Also, an x-y plotter can be driven, which will draw the spectrum on an appropriate graph. Similarly, data can be fed into an electric typewriter, which will print out channel number and counts in each channel. Or, information contained in a pulse-height analyzer can be put onto magnetic tapes, computer discs, or punched paper tapes. In these forms, data may be stored, or played through a computer, which can analyze the spectrum and automatically subtract out background to interpret the peak manner.

The curve was obtained (Figure 11-5-15) with a NaI(Tl) crystal, hence the poor resolution. E_1 is the energy of the gamma in the first peak; E_2 is the energy of the gamma in the second peak. Many modern pulse-height analyzers are provided with built-in computers to provide this capability.

Quite obviously, the more channels available, the more detailed the analysis of the spectrum can be. A 400-channel analyzer can look at 5-keV increments over a range from 0 to 2000 keV, but a 4000-channel unit can sort into 0.5-keV increments.

If desired, only a limited energy range may be analyzed. For example, most alpha particles emitted from a nucleus are in the 3.5 to 5.5 MeV region; only this region need be examined.

Figure 11-5-15.
Individual and composite spectra for two single-energy gamma-emmitting isotopes in a single sample.



Multichannel analyzers are equipped with a variety of electronic features that make them far more useful. For example, half of the channels may be used to count the unknown nuclide, and the information stored in the memory. The other half may be used to count a known nuclide. The two spectra may then be compared or overlaid on an oscilloscope display.

The key to spectroscopy of radionuclides is the ability of the instrument to resolve or separate pulses of different sizes. The multichannel analyzer can readily do this, but is limited by the detector. For example, a pulse-height spectrum (Figure 11-5-16) for ^{137}Cs may be obtained with NaI(Tl) scintillators of various sizes.

The ability of a detector to resolve or separate pulses is given by its FWHM (full width, half maximum) - the entire

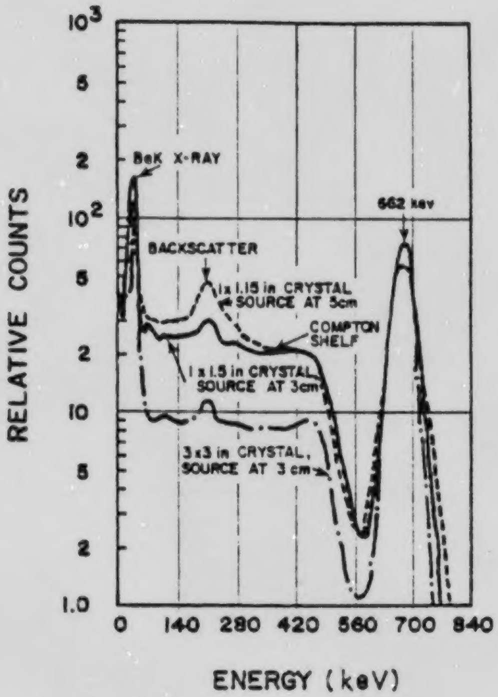


Figure 11-5-16.
^{137}Cs gamma spectrum obtained with three different size NaI(Tl) scintillation crystals. (Adapted from Price.)

width of the peak at one half the maximum counting rate (or counts) in a peak. For the 3x3 NaI(Tl) crystal (Figure 11-5-16), peak height is 800 counts and width at 400 counts (half maximum) is about 76 keV. Hence, its FWHM at 662 keV is 76 keV, or 76/662 = 11.5%.

At the low-energy end of the spectrum, the peak appears sharper, but resolution is actually poorer: FWHM is approximately 8.5 keV, which = at 35 keV (channel in which peak number of counts occurs) - gives a value of 24.3%.

Although NaI(Tl) and other scintillators have relatively poor resolution, they do have high efficiency for gamma rays because of their high Z and large size. Also, they are relatively inexpensive, rugged and can be decontaminated easily. And, unlike some semiconductors, they do not need cryogenic (cold) treatment.

Semiconductor detectors, although inefficient, have an ability to resolve different energy peaks (for example, the FWHM for ^{137}Cs peak at 662 keV of 1.5 keV, or only 0.22%. Also, at low energies, resolution is good (for example, at 5.9 keV, FWHM may be on the order of 0.2 keV, or about 3%).

Efficiency of a semiconductor detector, relative to NaI(Tl), may be only a few percent, but modern, large (and expensive) GeLi detectors may exceed 25% of the efficiency of a large NaI(Tl) crystal.

Another advantage of semiconductors, related to their resolving ability, is the reduced Compton shelf. From Figure 11-5-16, the peak-to-Compton ratio of about 3 is obtained with the small NaI(Tl) crystal, and 8 with the large one for ^{137}Cs. Typically, a GeLi detector will show a peak-to-Compton ratio of from 30 to 40.

A typical gamma spectra (Figure 11-5-17) may be obtained with either a GeLi or a NaI(Tl) detector on the same source, an excited state ^{166}Ho. The NaI(Tl) was unable to resolve the complex spectrum, although - as can be seen - its counting efficiency was very much greater, particularly for high energies. But the clear, sharp peaks of the GeLi detector virtually completely characterize the spectrum, thus making this detector far more valuable for nuclear physics studies or for identifying individual nuclides in mixtures.

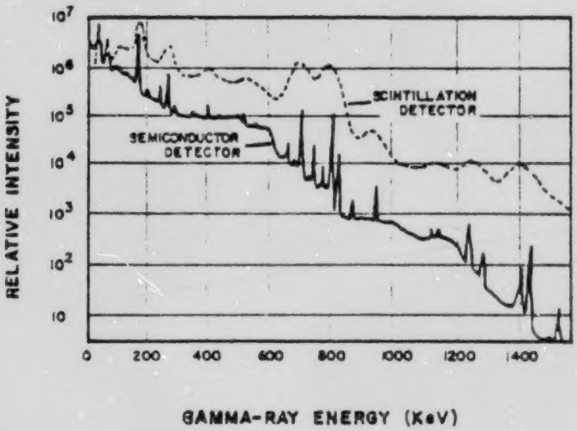


Figure 11-5-17.
Complex gamma ray spectrums of ^{166}Ho, obtained with NaI(Tl) scintillator and GeLi detector.
Note resolution of peaks and relative counts obtained.

Both junction and surface-barrier detectors are used for spectroscopy of charged particles. Laboratory identification of mixtures of alpha emitters are easily accomplished (Figure 11-5-18). Ion chambers of appropriate design are also used for this purpose.

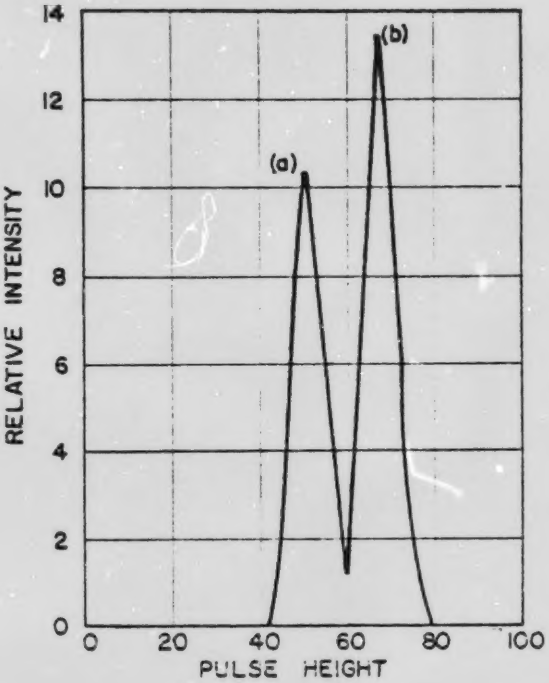


Figure 11-5-18. Differential pulse-height spectrum for a) 4.8-MeV alpha particles of ^{233}U, and b) 5.5-MeV alpha particles of ^{238}P obtained with semiconductor detector.

Beta spectroscopy is complicated by the fact that beta particles emitted from a nucleus have a whole spectrum of energies. Both semiconductors and scintillators have been used to accurately characterize beta spectra, with gas-filled detectors being used to a lesser extent.

The basic fundamentals of spectroscopy described above apply to alpha and beta spectroscopy as well as gamma spectroscopy.

EXERCISES

1. Compare various classes of scintillators, citing advantages and disadvantages of each.
2. Calculate true counting rate; use Equation 11-4-4 for a NaI(Tl) crystal, with an observed counting rate of $1.4x10^7$ cpm and a resolving time of 0.25 μsec.
3. Photomultiplier "A" has 10 dynodes with a secondary emission ratio of 4.5; photomultiplier "B" has 8 dynodes and a secondary emission ratio of 6.0. Which has a higher gain? To obtain the greatest gain, would photomultiplier "C" with 14 dynodes and a secondary emission ratio of 2.8 be your choice?
4. Calculate resolution of a detector that has the following characteristics for the 1.17 MeV ^{60}Co gamma:

    Peak Count Rate = 762 cps.
    Width of Peak at Half Maximum = 17 channels.
    PHA Calibration = 4 keV/channel.

    What type would you suspect this detector to be?

MATERIALS

1	Single-channel pulse-height analyzer.
1	Multi-channel pulse-height analyzer (optional).
1	Calibration set of gamma emitting nuclides.
1	Unknown mixture of two to four gamma emitters.
1	NaI(Tl) crystal.

1 GeLi detector (optional).
Graph paper.
1 Table of gamma emitters by energy.

PROCEDURES

SINGLE-CHANNEL PULSE-HEIGHT ANALYZER

1. Assign one group of two students a single-channel analyzer; instruct them to obtain one of the following energy spectrums:
   Background.
   Known gamma emitters - ^{137}Cs, ^{60}Co, ^{131}I (three groups, one source each).
   The remaining groups of two students should be given unknown samples, each containing different radioisotopes.
2. Number of intervals, counting time, starting and stopping positions and window widths should be given by instructor.
3. Take counts and record data on Data Table 11-5-1.
4. Each group should plot energy spectrums for all known emitters, cpm versus channel position. For convenience, use gross count rate.
5. Locate channel position for each calibration peak. By using energy of these peaks, plot energy versus channel position. (A straight line should result, which is the calibration curve for the pulse-height analyzer.)

6. Repeat Procedure 4, but use unknown emitters; identify peaks and assign the appropriate energies.
7. By using a table of gamma emitters arranged by energy, identify nuclides in unknown samples.

MULTI-CHANNEL PULSE-HEIGHT ANALYZER
(If Available)

1. Set up pulse-height analyzer with NaI(Tl) crystal as detector.
2. Obtain a 2-minute background count in half the channels and retain in memory.
3. Count each gamma calibration source for 2 minutes in the other half of the analyzer; use analyzer to subtract out background.
4. Locate channel number for each calibration peak; plot energy versus channel number on graph paper. (A straight line should result, which is the calibration curve for the pulse-height analyzer.)
5. Repeat above procedure with an unknown mixture of gamma emitters; identify peaks and assign appropriate energies.
6. By using a table of gamma emitters arranged by energy, identify nuclides in mixture.
7. Refer to data obtained with calibration sources; compute efficiency of NaI(Tl) crystal for various energies; plot results on log-log paper; use this plot to estimate amount of each nuclide in unknown mixture. Note: Begin with highest energy peak; strip it out of spectrum along with its expected Compton contribution

and escape peak.

8. Repeat above procedure with a GeLi detector (if available).
9. Which detector was easier to use? Why?
10. Which detector gives more accurate results? Why?

DATA TABLE 11-5-1

Type of count (e.g., background, known, unknown): __________

Type and activity of emitter, if known: ____________________

Counting times: ____________________

Starting position: ____________________

Stopping position: ____________________

Window width: ____________________

No. of intervals: ____________________

Energy Position (Threshold)	No. of Counts	Gross Count Rate

REFERENCE MATERIALS

Chase, G.D. and Rabinowitz, J.L. *Principles of Radioisotope Methodology*. Minneapolis, MN: Burgess Publishing Co., 1965.

Faires, R.A. and Parkes, B.H. *Radioisotope Laboratory Techniques*. 3rd ed. New York: Halsted Press Division, John Wiley and Sons, 1973.

Glasstone, Samuel, ed. *Sourcebook on Atomic Energy*. 3rd ed. New York: Van Nostrand-Reinhold, 1967.

Hine, Gerald J. *Instrumentation in Nuclear Medicine*. New York: Academic Press, 1967.

International Commission on Radiological Units and Measurements (ICRU). "Measurement of Low Level Radioactivity" Report No. 22. Washington, DC: National Bureau of Standards, 1972.

Mann, W.B. and Garfinkel, S.B. *Radioactivity and Its Measurement*. New York: D. Van Nostrand, 1966.

"Nuclear Spectroscopy Instrumentation." *Nuclear Instruments and Methods* 43 (1966):1-277.

Quimby, E.H. and Feitelberg, S. *Radioactive Isotopes in Medicine and Biology*, Vol. 1, Chapter 12. New York: Lea and Febiger, 1965.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. What are scintillations?

2. What is luminescence? How does luminescence differ from phosphorescence?

3. Answer these questions about inorganic crystal scintillators.

    a. Briefly describe the inorganic crystal detector's principle of operation.

    b. For what type(s) of radiation can this detector be used?

    c. What are four advantages to using sodium iodide as the organic material?

    d. Why is a small amount of thallium added to the sodium iodide inorganic crystals?

4. Answer these questions about organic crystals:

   a. Describe the luminescence process in organic crystal scintillators.

      ____

      ____

      ____

      ____

   b. For what type(s) of radiation are organic crystal scintillators used?

      ____

   c. Name two commonly used organic crystals. ____________________

      ____

5. Answer these questions about liquid scintillators:

   a. Why are liquid scintillators more sensitive than either inorganic or organic crystals? ____________________

      ____

   b. Describe the mechanism of interaction with radiation to produce light.

      ____

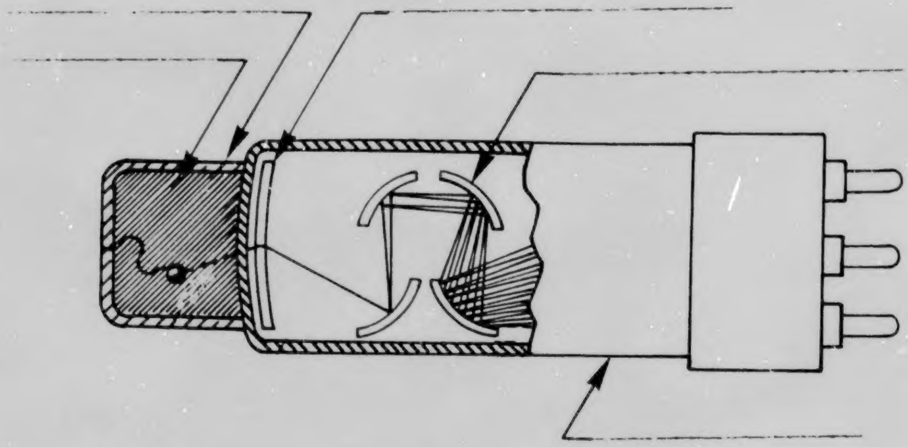
      ____

      ____

   c. What major advantage does a liquid scintillator have over other scintillators? ____________________

      ____

6. Label the parts of the scintillation phosphor and photomultiplier tube to which the arrows point.



7. What are the main purposes of the photomultiplier tube?

8. Define these terms as they apply to scintillation counting.

   a. Photocathode

   b. Dynodes

   c. Secondary Emission Ratio

9. Calculate the total gain of a photomultiplier tube with 10 stages and a secondary emission ratio of 6 to 3.

10. Draw a simplified block diagram of either a sodium iodide scintillation counting system or of a single channel liquid scintillation system that uses coincidence logic.

11. What is the purpose of the well in the crystals used in scintillators?

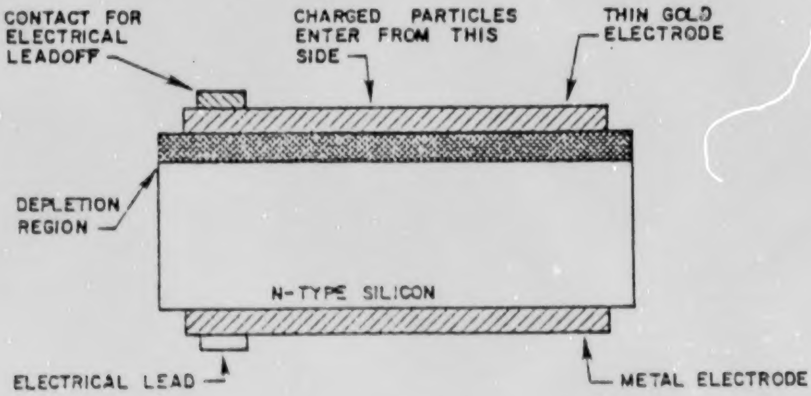
12. What is a semiconductor? What is its main application?

13. Briefly describe the operation of a diffused p-n junction detector.

14. What is meant by the terms " p materials" and "n materials"? How are they formed?

15. What causes the depletion layer to be formed in the semiconductor detector?

16. Below is a schematic drawing of a surface-barrier detector. Use the drawing to answer parts a-d of this question.


CONTACT FOR ELECTRICAL LEADOFF
CHARGED PARTICLES ENTER FROM THIS SIDE
THIN GOLD ELECTRODE
DEPLETION REGION
N-TYPE SILICON
ELECTRICAL LEAD
METAL ELECTRODE

a. In n-type crystals, such as silicon or germanium, why are free electrons repelled?

b. What purpose does the gold electrode serve?

c. What is the purpose of the metal electrode?

16. d. This detector has a thinner depletion region than the diffused p-n junction. What, if any, significance does this have?

17. Briefly discuss three major advantages and disadvantages surface barrier and junction-type detectors have over gas-filled detectors and scintillators.

18. In the lithium-drifted semiconductor, what is the purpose of silicon or germanium? Of lithium?

19. What is a Geli-detector? How does its stopping power and efficiency compare to a lithium-drifted silicon detector?

20. Briefly describe how the process of spectroscopy is carried out.

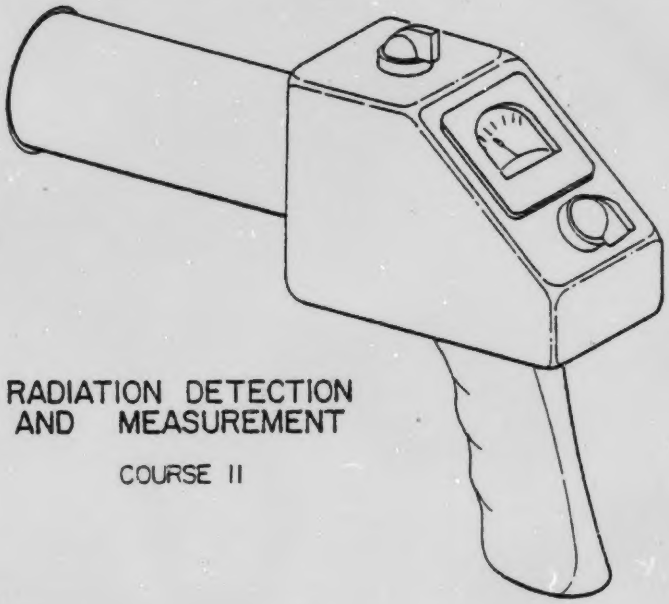
21. For what types of radiation is spectroscopy useful?

22. Draw a block diagram of a spectroscopy system.

23. What is pulse height analysis?

NUCLEAR TECHNOLOGY

MODULE 11-6 EXTERNAL PERSONNEL DOSIMETERS FOR BETA-PHOTON MONITORING



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
⇨ MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

Previous modules in this course have been primarily concerned with detection and measurement of radiation in terms of individual ionizing events rather than in terms of dose. This module, however, is concerned with dosimeters - devices that read out or are calibrated in units of dose.

There are many kinds of dosimeters; those designed to be worn by workers in radiation areas are known as "personnel dosimeters." Other than pocket ionization chambers (discussed in Module 11-2, "Ionization Detectors"), most personnel dosimeters are small, solid-state devices, worn on the body of the radiation worker.

After an appropriate time interval, usually a month, a dosimeter is collected and processed, and the results evaluated by a health physicist. Thus, a continuous record of radiation exposure of every employee in a nuclear power plant is obtained, with the information used to control and limit the extent of individual radiation exposure.

In addition such information acts to ensure that day-to-day safety precautions and rules have been followed and are adequate. No other occupational hazard is so well controlled or monitored, even though many toxic dusts, fumes, gases, mists and vapors may present significant hazards to the health of workers.

This module will review radiation units and permissible dose levels and provide a discussion of various types of personnel dosimeters in general use.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe fundamental principles of responses of photographic and luminescent dosimeters to ionizing radiation.
2. Discuss limitations of the three major personnel dosimetry systems used for β-γ monitoring.
3. Calibrate TLD's and read out known exposures.
4. Select dosimeters appropriate to given exposure situations.
5. Define or describe the following terms:
    - Roentgen.
    - Rad.
    - Rem.
    - Quality factor (QF).
    - Latent image.
    - Sensitivity.
    - Energy dependence.
    - Radiophotoluminescence.
    - Thermoluminescence.
    - Glow curve.
    - Glow peak.

6. Give the composition of RPL glass.
7. Name three TLD materials.
8. Differentiate between absorbed dose and biological dose.

RADIATION UNITS AND PERMISSIBLE LEVELS

Radiation measurements and units of dose are based primarily on the energy deposited by the radiation as it travels through matter. Shortly after the discovery of x-rays and radioactivity, attempts were made to quantify (measure) radiation by various color changes induced in chemicals, by counting scintillations or by measuring the amount of blackening in a photographic plate that had been exposed to radiation. A few years later, it was thought that ionization of air would be the most accurate and suitable way of measuring the radiation intensity. But by 1920, a quarter of a century after the discovery of x-rays and radioactivity, there was still no commonly accepted (or used) unit of radiation dose or exposure.

Since the mid-1920's, the basic unit has been the roentgen (R), which has long been associated with x-ray measurement. The current definition of a roentgen is "that quantity of x or gamma radiation that produces 2.58 x 10^{-4} coulombs in 1 kg of air," which corresponds to an energy deposition of about 88 ergs/g of air.

Recall that in Module 11-2, "Gas-Filled Detectors I: Ionization Detectors," W was defined as "the average energy needed to produce one ion pair and is equal to 34 eV/ion pair." The following calculation shows this relationship:

$$1\ \text{R} = 2.58 \times 10^{-4}\ \frac{\text{coulomb}}{\text{kg}} \times 6.28 \times 10^{18}\ \frac{\text{charges}}{\text{coulomb}} \times \frac{34\ \text{eV}}{\text{ion pair}}$$

$$\times\ 1.6 \times 10^{-12}\ \frac{\text{erg}}{\text{eV}} \times \frac{0.001\ \text{kg}}{\text{gram}}$$

$$= 88\ \text{erg/g}.$$

Note that the definition of a roentgen has two major limitations. First, it applies only to air and, second, it applies only to x or gamma radiation; therefore, it is not correct to refer to a person's dose in terms of roentgens. The roentgen itself is considered to be the unit of exposure, with exposure in this case defined in a very restricted sense, namely the amount of electric charge imparted to a kilogram of air by x or gamma radiation.

Since the definition of the roentgen is rigidly limited to photons in air, another unit must be used for other kinds of radiation and other absorbing media. A unit known as the rad* has been defined in terms of the energy absorbed from the ionizing radiation per unit mass of the absorbing medium. One rad is equal to 100 ergs per gram; or, in the new SI units, one rad is equal to 0.01 joules (J) per kilogram. The rad can be used to specify the absorbed dose from any kind of ionizing radiation in any medium. Note that for all practical purposes, 1 R ≈ 1 rad.

While the rad is a useful unit, it should be noted that different radiations may exert different biological effects, even though they produce the same dose in tissue; therefore, biological response to the radiation must be taken into account to express doses in terms of biological effects. This unit is called a rem - a unit that produces the same damage to humans as 1 R of high voltage x-ray. A dose equivalent (DE) expressed in units of rem is given by Equation 11-6-1.

---

* Recently, the international scientific bodies have created a new unit, the gray (Gy), equal to 1 J/kg, which is to be the new unit of absorbed dose in place of the rad; however, the rad will continue to be used on an interim basis and will, of course, appear in all earlier publications.

DE = D x QF x DF x . . . Equation 11-6-1
(Equation = Eq.)

in which

DE = dose equivalent in units of rem.
D = absorbed dose in units of rad.
QF = quality factor (which will be defined later).
DF = distribution factor.
... = any other factor that modifies or determines the response of a particular person to the radiation.

On practical grounds, the DF and the . . . (called an "ellipse") can be eliminated. The DF is used only for internal doses where a radioactive substance may be lodged in one particular organ, exposing it to a greater dose than the rest of the body. The other factors are considerations relating to the age or health of the individual and are not commonly applied nor can they be measured by a dosimeter. Thus, for practical purposes, DE = D x QF.

The quality factor (QF) is a somewhat arbitrary measure of the effectiveness of the various types of radiations at producing biological damage. It is related to the linear energy transfer or specific ionization of the radiation, as shown in Table 11-6-1. The linear energy transfer, or LET, is expressed in keV/μm and is the amount of energy expended per μm of track length by the ionizing radiation. Thus, alpha particles, with their dense ionizing tracks, would be expected to have a greater QF than a gamma ray. Table 11-6-2 summarizes the QF's used for radiation protection work for different kinds of radiation, and

TABLE 11-6-1
RELATIONSHIP OF QUALITY FACTOR TO LINEAR ENERGY TRANSFER

LET in keV/μm	QF
≤ 3.5	1
> 3.5 - 7.0	2
> 7.0 to 23	5
> 23 to 53	10
> 53 to 175	20

TABLE 11-6-2
QUALITY FACTORS IN PRACTICAL USAGE

Radiation	QF
Photons, electrons, or positrons (E > 0.03 MeV)	1
Neutrons (E < 10 keV)	3
Neutrons (E ≥ 10 keV)	10
Protons	1 to 10
Alpha particles	1 to 20
Fission fragments	20

Table 11-6-3 summarizes radiation units.*

TABLE 11-6-3
SUMMARY OF RADIATION UNITS

Unit	Symbol	Application	Comments
Roentgen	R	Exposure	Limited to photons in air = 2.58×10^{-4} J/kg
Rad	rad	Absorbed dose	= 100 ergs/g = (10^{-2} J/kg)
Gray	Gy	Exposure	= 1 J/kg
Rem	rem	Radiation protection	= D(rad) x QF x DF...
Seivert	Sv	Radiation protection	= D(Gy) x QF x DF...
Quality factor	QF	Radiation protection	Function of LET
Distribution factor	DF	Internal dose	Primarily useful for organ dose calculations
Linear energy transfer	LET	Quality-factor determination	= keV/μm
Curie	Ci	Activity	= $3.7 \times 10^{10} \frac{\text{disintegrations}}{\text{sec}}$

It should be noted that dose limits (as given in Table 11-6-4) are given in units of rem, but no dosimeter responds in a rem or biological dose fashion. Dosimeters can be made to approximate the response of the body to radiation,

---

* As is the case with the unit for absorbed dose, a new unit for biological dose has been defined by the international bodies. This is the Seivert (Sv), which is equal to the dose in Gy x QF and other appropriate modifying factors.

TABLE 11-6-4
RECOMMENDED ANNUAL DOSE-EQUIVALENT LIMITS

Occupational Exposure	
Total body, head and trunk, blood-forming organs (red bone marrow), gonads and lens of eye	3 rems in any calendar quarter but not to exceed 5(N-18) rems, where N is the age of the individual in years
Once in a lifetime-accidental exposure	25 rems
Skin of whole body, thyroid and bone	15 rems/year
Hands, forearms, feet and ankles	75 rems/year
Other internal organs	15 rems/year
Individuals Not Directly Engaged in Radiation Work	
Whole body, gonads and blood-forming organs	1.5 rems/year
Other organs, tissues and extremities	One-tenth of the corresponding occupational dose
Population at Large	
Whole body, gonads and blood-forming organs	0.5 rem/year to an average maximum per individual of 0.17 rem/year for the whole population
Other organs	1.5 rems/year to an average maximum per individual of 0.5 rem/year

but cannot be made to duplicate the often subtle and individual variable responses of living tissue to ionizing radiations.

Similarly, it is a practical impossibility to make a dosimeter that is truly tissue equivalent. Dosimeters can be made that approximate soft tissue, bone or specific organs (such as air-filled lungs), but making a dosimeter that incorporates all of these elements is impossible. And, because of density, different Z and other factors, the dose to bone, for example, can be very much different from the dose to soft tissue overlying it. Thus, personnel dosimeters are designed to measure the dose at the surface of the body, where it is usually the greatest, or at a depth of 1 cm in soft tissue, approximately the depth of the blood forming organs.

One additional unit bears discussion: the curie. The curie is the unit of activity exactly equal to 3.7 x 10^{10} nuclear disintegrations per second. The curie is related to dose in a complex way, much as a gallon of gasoline is related to how far one can travel in a motor vehicle: if the vehicle is a heavy truck or bus, the distance may be only a few miles, while a lightweight motorcycle might go more than 100 miles. The dose is determined by the energy and the kind of radiation more than by the quantity of radioactivity.

Working with radiation, as with any other form of human endeavor, carries with it some risk. To minimize this risk, various organizations have established maximum permissible dose limits (summarized in Table 11-6-4) which are shown to indicate the magnitude of the doses and ranges that personnel dosimeters may be required to measure.

Remember, personnel dosimeters measure doses from radiations "external" to the body that can penetrate the skin and produce a dose to living tissue. Alpha particles, with their very limited range, do not constitute an external hazard since they cannot penetrate the dead external layers of the skin. The same is true for protons and fission fragments. Therefore, personnel dosimeters are needed only for monitoring dose from β rays, photons (gamma and x-rays) and neutrons.

PHOTOGRAPHIC FILM DOSIMETERS

Photographic films are the oldest and most widely used personnel dosimeters. As long ago as 1903, an American physician, Dr. E. Stern, used photographic emulsions to monitor the x-ray doses his patients were receiving, and in the late 1920's, a physicist, Dr. Edith Quimby, devised the first film dosimeter.

A photographic emulsion is a mixture of tiny crystals of silver bromide suspended in gelatin, spread in a layer across a sheet of thin plastic used for support and rigidity. The sensitivity of the photographic emulsion depends upon many factors, including the type and energy of the radiation, the amount of silver bromide, the size and shape of the crystals or grains and the types of impurities introduced into the crystals or the emulsion.

Beta particles, photons and other radiation passing through the emulsion impart a small amount of energy to the silver bromide crystals, causing some of the silver in the crystal to be reduced from the ionic state to free atomic silver. These atomic silver atoms are known as the latent image, which is amplified by subsequent chemical treatment or development. After development, the atomic silver remains behind on the film base and appears black in color. The amount of blackening is a measure of the radiation exposure and is determined as the "optical density," an inverse measure of light transmission through the developed film.

The optical density of an exposed and developed photographic film is a function of the radiation dose (Figure 11-6-1). This characteristic curve for the response of

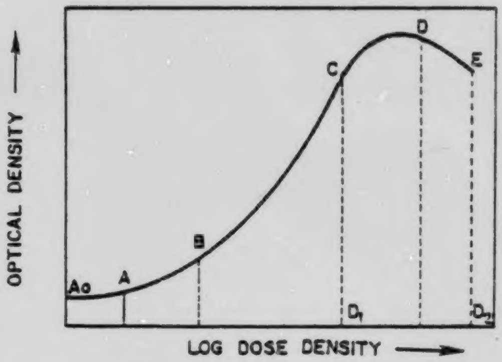


Figure 11-6-1.
Characteristic response curve of a photographic film exposed to ionizing radiation. (Labels are described in text.)

a photographic emulsion consists of five smoothly joined segments. In the first, from A_0 to A, the density does not appreciably increase with the dose, and the film is not usable in this region. From A to B, the density is proportional to dose, and the film becomes useful for dosimetry. The region AB is known as the toe of the curve. The next region, from B to C, is the largest and, in this region (where the film is most accurate), the density is proportional to the logarithm of the dose.

The region bounded by CD is the shoulder of the curve, where density hardly increases, reaching a peak and then beginning to drop off as shown in the tail from D to E. The reason for the drop is complex and the phenomenon is called "solarization."

If solarization occurs, the density reading at S_o could be interpreted as being a dose of D_1 or D_2; therefore, films must be carefully interpreted to avoid errors caused by solarization. One way of minimizing such errors is to include a second piece of film with a broader range that overlaps. Another way is to use a film with low solarization potential.

A skilled film interpreter can generally identify solarization by other response characteristics and, since doses are usually quite high, an investigation by health physics personnel can also shed light on the exposure.

The density or light transmission of the film is determined by a "densitometer." The densitometer has a strong, focused light source directed onto a photocell. With no film in place, light transmission is 100%, but with film inserted between the light source and the photocell, transmission will be reduced. Even an unexposed or clear film will reduce light transmission in some degree. The optical density, usually symbolized by O.D. or S, is defined as follows:

$$\text{O.D.} = \log \frac{I_o}{I} . \qquad \text{Eq. 11-6-2}$$

where

I_o = light incident on the film (or the reading with no film in place).

I = amount of light transmitted through the film.

Since even an unexposed film will show some light absorption or "background density," this quantity must be subtracted out to obtain the "net optical density" (N.O.D.), which is the amount of blackening attributable to the radiation dose. The N.O.D. is calculated by subtracting out the background density, I_{Bkgd}, as shown in Equation 11-6-3:

$$\text{N.O.D.} = \log \frac{I_o}{I} - \log \frac{I_o}{I_{Bkgd}} . \qquad \text{Eq. 11-6-3}$$

Or, by performing the indicated subtraction:

$$\text{N.O.D.} = \log \frac{I_{Bkgd}}{I} . \qquad \text{Eq. 11-6-4}$$

As shown by Equation 11-6-4, only the background or control film density and the dosed (unknown) film density need be obtained.

Most densitometers are equipped with a three-decade logarithmic meter that permits readout from about 0.01 to 3 or 4, which is the upper range of most instruments. A few may read to a density of 6, but these are quite expensive and usually used for research rather than routine personnel dosimetry. Developed unexposed film will typically read a few hundredths to a few tenths of a density unit.

The sensitivity of the film is how much blackening or optical density is produced by a given radiation exposure. Sensitivity has a precise physical definition, namely the reciprocal of the exposure required to produce a density of 0.30. Units of sensitivity are, therefore, $\frac{1}{R}$ or R-1. The sensitivity of a photographic film can be controlled in many ways, chiefly by varying the grain size.

A larger grain size will result in a more sensitive film, since only one reduced silver atom in a grain (the latent image) is needed to cause development of the entire grain. And, the larger the grain size, the greater the probability that a given dose will produce a latent image of one or more reduced Ag atoms in it.

Sensitivity is also regulated by emulsion thickness and by the addition of certain sensitizing agents to the emulsion or silver bromide grain, or by coating the film base on one or both sides.

Typically, photographic films are designed to have a sensitivity of about 5 R^{-1}; that is, an exposure of about 0.2 R will produce a density of 0.3. This provides a useful range of about 100 mR to perhaps a few R (Figure 11-6-2). A second film, covering the range from about 1 R to perhaps 800 R, may also be included.

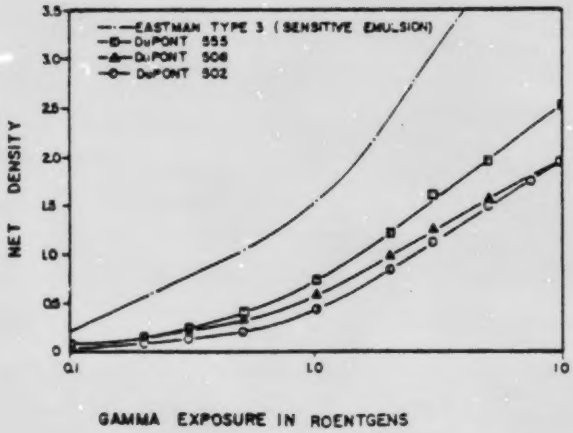


Figure 11-6-2. Density versus dose relationships for four personnel dosimetry films.

Sometimes, a film base may be coated on one side with a high-sensitivity emulsion and on the other with a low-sensitivity emulsion, combining both features and a very wide range into one piece of film. The high-range (i.e., the low-sensitivity) emulsion is ready after first stripping off the darkened low-range emulsion from the film base.

There are many drawbacks and difficulties to the use of films, one being the phenomenon of energy dependence. Energy dependence is a result of the high Z components in film, primarily the silver ($Z = 48$) and bromine ($Z = 35$) that make up the sensitive portion of the emulsion.

The photoelectric effect predominates at low energies as the mechanism of photon interaction with matter. The probability of the photoelectric effect is proportional to Z^5; the high effectiveness of Z of the emulsion (relative to air: $Z_{eff} = 8.14$) produces a pronounced overresponse for photons with energies below about 400 keV (Figure 11-6-3). The density produced by 1 R of 30 keV photons could be 30 to 40 times the density produced by 1 R of photons (gamma rays) from ^{60}Co, with an effective energy of 1250 keV.

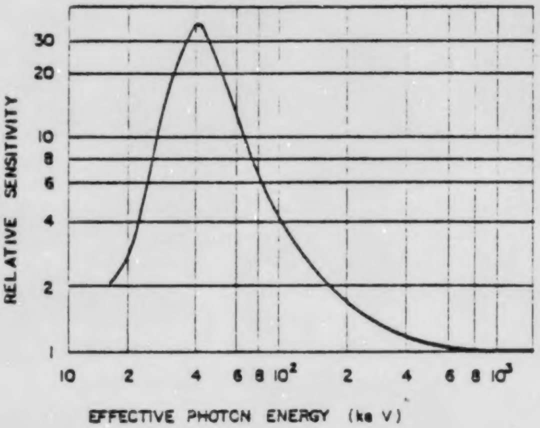


Figure 11-6-3.
Typical energy dependence curve for a personnel monitoring film.

To minimize the effects of energy dependence, metal filters can be used to absorb the lower energies and allow the higher energies to pass. The filters generally should be matched to the particular type of film used. Ordinarily, a single high-Z filter can flatten the response per unit exposure over a wide range; e.g., 50 keV to several MeV (Figure 11-6-4). Other filters can be used to discriminate against β rays and to evaluate the lower energy photons. Ideally, four filters should be sufficient for all cases as shown below:

Filter	Radiation Passed
High Z filter	Higher energy photons, no low energies.
Medium Z filter	Photons with some attenuation of low energies.
Low Z	Photons with no attenuation.
None (open window)	β + photons with no attenuation.

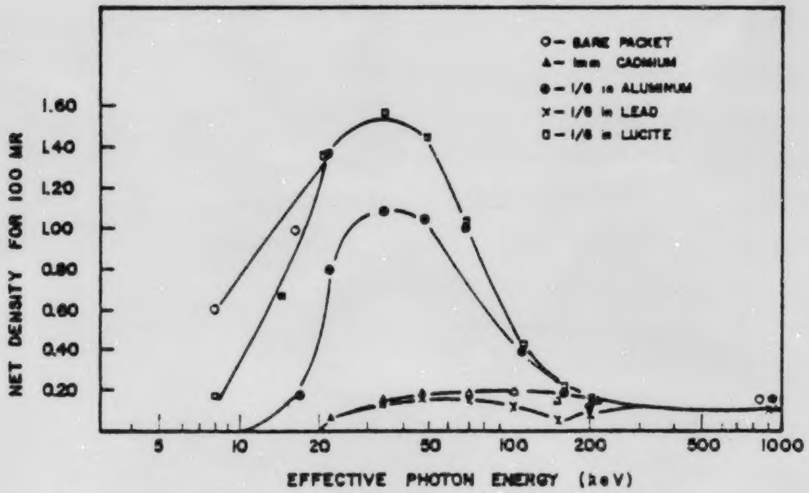


Figure 11-6-4. Photon energy response of film (DuPont 502) in badge through indicated filters.

The low Z filter reading is subtracted from the open window to obtain the β dose contribution. The density under the high Z filter gives the dose from higher energy photons. The ratio of low Z to medium Z density readings gives an indication of the average energy of the remaining portion of the photon dose, and the proper dose estimate can be made based on knowledge of the energy dependence of the particular film used.

The upper badge (Figure 11-6-5) is designed for use

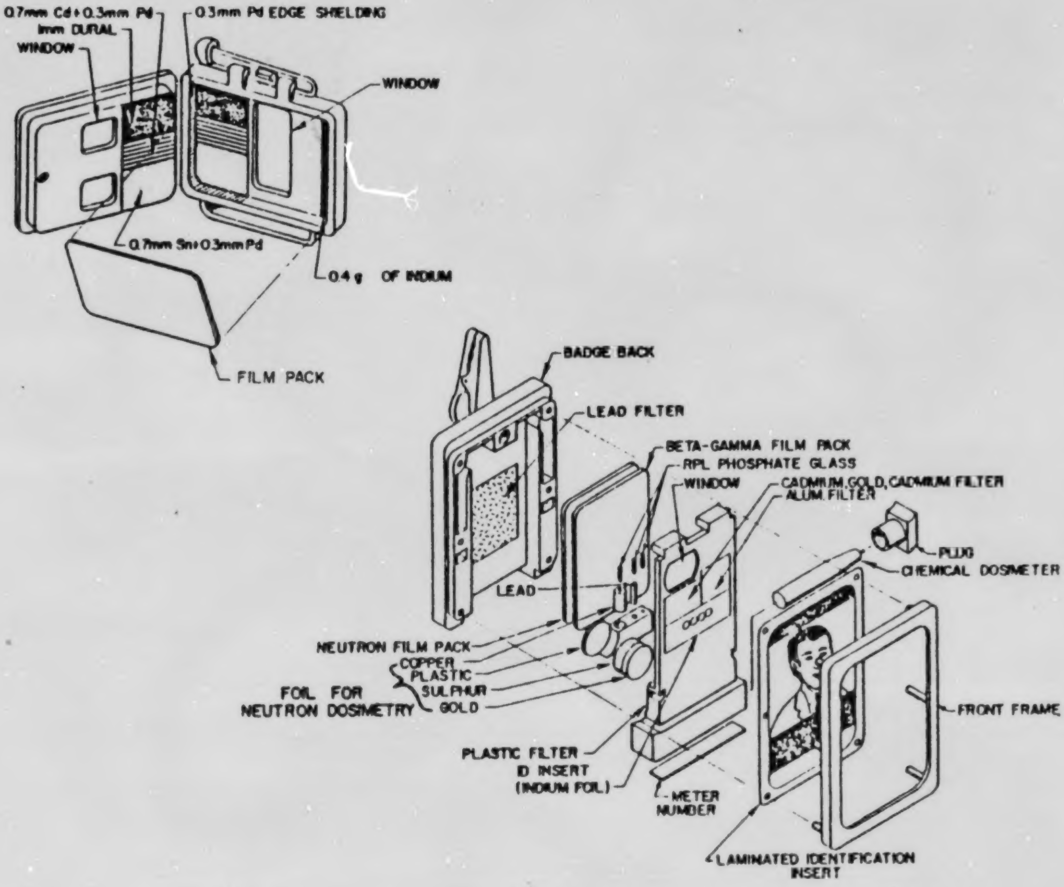

0.7mm Cd+0.3mm Pd
0.3mm Pd EDGE SHIELDING
1mm DURAL
WINDOW
WINDOW
0.7mm Sn+0.3mm Pd
0.4 g OF INDIUM
FILM PACK
BADGE BACK
LEAD FILTER
BETA-GAMMA FILM PACK
RPL PHOSPHATE GLASS
WINDOW
CADMIUM,GOLD,CADMIUM FILTER
ALUM. FILTER
PLUG
CHEMICAL DOSIMETER
LEAD
NEUTRON FILM PACK
COPPER
PLASTIC
SULPHUR
GOLD
FOIL FOR NEUTRON DOSIMETRY
FRONT FRAME
PLASTIC FILTER
ID INSERT (INDIUM FOIL)
METER NUMBER
LAMINATED IDENTIFICATION INSERT

Figure 11-6-5. Two examples of personnel film dosimeters.

with a single film packet and clips to the clothing with a simple but effective safety pin. The other, used in a different installation, is more complex, containing neutron monitoring film, foils and radiophotoluminescent glass rods made luminescent by irradiation with β and γ rays followed by exposure to light.

The dose interpretation of exposed personnel film is complex and requires skilled personnel. Calibration curves for different photon energies are obtained and the data from these can be used to interpret the film. A series of four simultaneous equations is obtained, and solution can be achieved with the aid of a computer.

Despite the use of filters and sophisticated mathematical techniques, the problem of energy dependence plus the inherent variability in films makes photographic film dosimetry relatively inaccurate, particularly for mixed radiations. For low doses of mixed radiations, errors of ± 50% to ± 200% are not unusual; however, ± 25% can be achieved in most personnel dosimetry situations.

To some extent, the errors may be a result of other characteristics of the photographic films. Developing must be rigidly controlled (time, temperature and developer composition can affect the development process). Because films are made in the shape of thin, flat sheets, they are angular dependent, with an edgewise exposure producing a different response per unit exposure than one perpendicular to the plane of the film. And, films are significantly affected by their environment, including temperature and humidity (Figure 11-6-6).

Environmental conditions may alter the response of the film or affect the stability of the latent image. Under normal conditions, the latent image may last for a few

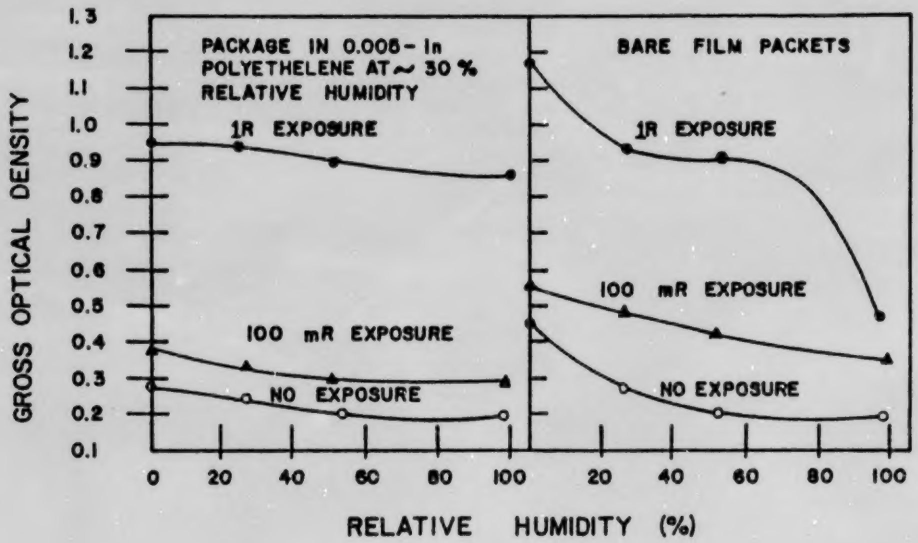


Figure 11-6-6. Gross optical density of personnel monitoring films held in air at 20°C for 13 weeks after exposure.

months, and then begin to fade. Thus, a film exposed to 1 R may show a density of 1.0 if developed, for example, within twelve weeks after exposure. The density produced by the same exposure will be smaller after this twelve-week period. Latent image fading is strongly influenced by relative humidity (Figure 11-6-6).

The response of photographic films to radiation may also be directly affected by temperature, as well as by humidity. At high temperatures, above about 60°C (140°F), fogging or darkening due to heat may occur. Chemicals such as mercury or other air pollutants may also diffuse through the film wrapper, altering the response or adversely affecting the latent image. Films must be kept in light-tight packages, for they are extremely sensitive to light; even a pinhole leak can ruin a film. Pressure static charges build up on the film, and contamination also can cause spurious readings or ruin the film.

Table 11-6-5 summarizes the characteristics of film in terms of the ideal dosimeter. Note that film can be used for personnel monitoring of neutrons, as well as betas and photons. (This application is discussed in Module 11-7, "Neutron Detectors, Dosimetry and Activation Analysis.")

TABLE 11-6-5

COMPARISON OF THREE PERSONNEL DOSIMETERS IN TERMS OF THE "IDEAL PERSONNEL DOSIMETER"

Characteristic	Photographic Film	Radiophoto-luminescent Glass	Thermo-luminescence
Accuracy in the field	± 25%	± 10%	± 10%
Energy dependence	High, factor of 30 at 30 keV	Moderate, factor of 8 to 10 at 30 keV	Low, ± 25% from 0.01 to 1.25 MeV (LiF)
Angular dependence	High	Moderate	Low
Range	10 mR to 10^3R	20 mR to 500 R*	10 mR to 10^5 R*
Mixed radiations	Good	Poor	Fair
Cost	Low	Moderate	Low
Small size and convenient shape	Fair	Fair	Very good
Ease of handling and ruggedness	Fair	Good	Very good
Adaptability	Poor	Fair	Excellent
Longevity of wearing interval	1 Month	Indefinite	Indefinite
Shelf life	1 year	Years	Indefinite
Resistance to environment	Poor	Excellent	Excellent
Readout (includes rapidity, capability for subsequent readout, or reuse)	Fair	Very good	Good

* May be extended to 10^6 or more by proper dosimeter selection.

RADIOPHOTOLUMINESCENT GLASSES

Radiophotoluminescent glasses are another type of solid-state dosimeter with limited application to personnel monitoring. They are, however, widely used in research with experimental animals since they are small, chemically inert, and can be implanted directly into tissues to measure the dose at that point. Except for high-level or accident dosimeters, radiophotoluminescent (RPL) glass is hardly used as a personnel monitor in the United States. But in Europe (primarily Germany) and Japan, RPL glasses are more widely used for routine low-level personnel dosimetry.

Radiophotoluminescent dosimeters are generally silver-activated phosphate glasses. A typical glass contains 50% by weight each $LiPO_3$ and $Ag(PO_3)_3$, with the addition of 8% by weight $AgPO_3$ (silver metaphosphate) and 3% B_2O_3. When exposed to ionizing radiation, stable and permanent fluorescing centers are formed within the glass matrix. When exposed to ultraviolet light, these centers become excited and emit an orange fluorescence. The amount of the orange fluorescence is directly proportional to the prior radiation dose of the glass.

In an appropriate reader for RPL glass (Figure 11-6-7), the orange light is passed through an optical filter, which removes any scattered ultraviolet or other extraneous light, and a photomultiplier tube is used to amplify the signal. The whole light system is contained within a light-tight housing to avoid unwanted readings from visible light.

Since all RPL glasses contain silver ($Z = 48$) as the active agent, they exhibit some energy dependence. An appropriate filter can produce a flat response from as low

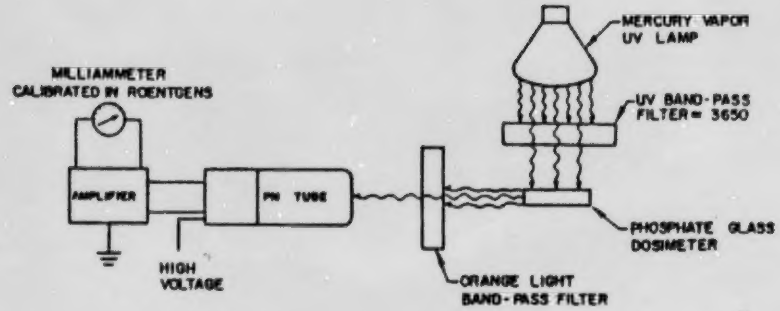

MERCURY VAPOR UV LAMP
UV BAND-PASS FILTER≈ 3650
MILLIAMMETER CALIBRATED IN ROENTGENS
AMPLIFIER
PM TUBE
PHOSPHATE GLASS DOSIMETER
HIGH VOLTAGE
ORANGE LIGHT BAND-PASS FILTER

Figure 11-6-7. Schematic representation of a reader for radiophotoluminescent glass.

as 80 keV (or perhaps even lower) to several MeV for photons. Two or three glasses can be used with different filters to obtain mixed β photon doses, much in the same manner as is done with film.

RPL glass, as a dosimeter, is superior in many ways to photographic film. First, its response (fluorescent light output per unit exposure) is linear rather than logarithmic and may remain linear over a very broad range. Typically, a small glass block, 8 x 8 x 4.7 mm, will have a useful range of about 30 mR to 300 R. When combined with a small glass rod, 1 mm in diameter by 6 mm long, the range can be extended to 10^5 R or even higher - well beyond the personnel dosimetry or even accident dosimetry range. The rods are useful down to a few R, providing a wide overlap with the useful range of the blocks. The overlap with film is not nearly this broad; in fact, it may be almost absent, contributing to the difficulty of interpretation and errors in the 1 to 10 R region.

Second, RPL glass is virtually free of environmental effects. The temperature during readout has very slight effect, generally less than 1% per °C. Humidity and chemicals have no effect on either response or readout. Dirt on the glass, however, can result in spurious fluorescence, and lengthy exposure to sunlight can cause a small reduction in the amount of fluorescence per unit of exposure. Readout is fast, easy and nondestructive; that is, the glass can be read out an infinite number of times. And, after readout, the RPL glass dosimeter can be returned to the field for additional dose accumulation, or it can be stored to provide a permanent re-readable record of the dose to that particular personnel dosimeter.

There appear to be only three drawbacks to the use of RPL glass: 1) the energy dependence, discussed above, requires several pieces of glass to obtain the correct dose in a mixed beta-photon field; 2) the glass has a relatively high cost and the need to use several pieces increases the cost; 3) handling is inconvenient and must be accomplished with forceps to avoid transferring dirt or grease from fingers onto the glass.

In a personnel monitor containing RPL glass (Figure 11-6-8), the external shield with conical holes is designed

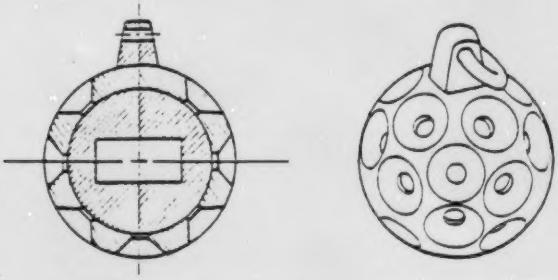


Figure 11-6-8. Dosimeter using RPL glass. (Casing with conical holes is designed to reduce energy dependence. Entire unit is about 1 inch in diameter.)

to minimize energy dependency. (NOTE: Characteristics of RPL glass in terms of the ideal personnel dosimeter were given in Table 11-6-5.)

THERMOLUMINESCENCE

Thermoluminescent dosimeters (TLD's) are rapidly replacing film badges in the United States. Several commercial firms offer such systems, and many large nuclear facilities, including the DOE Hanford site, Lawrence Livermore Laboratory and the Savannah River Plant, currently have their own internal TLD systems for personnel monitoring. Through 1975, an estimated 50% of all radiation workers issued personnel dosimeters were badged with TLD's.

Thermoluminescence is a property of all inorganic crystals. When exposed to radiation, a few electrons are raised in energy level from the valence band to the conduction band of the crystal. Most of these quickly drop back into the valence band, but some are trapped at imperfections in the crystalline lattice of the crystal. These imperfections may be strain points, cleavage planes or, more commonly, imperfections caused by impurities within the crystal.

When an electron is trapped between the valence and conduction bands, it cannot escape unless some additional energy is supplied. Thus, when the crystal is heated, the trapped electrons are raised back into the conduction band and quickly fall back into the valence band, emitting their excess energy as visible light. The imperfections thus form electron traps at various depths or energy levels within the forbidden zone between the valence and conduction bands. More heat is required to remove the electrons from the deeper trap; as the crystal is heated, light pulses occur at various temperatures corresponding to different trap depths.

A solid-state diagram illustrating the thermoluminescence phenomenon (Figure 11-6-9) and a glow curve (Figure 11-6-10) show the emission of light at different temperatures. The curve shows five significant and specific peaks

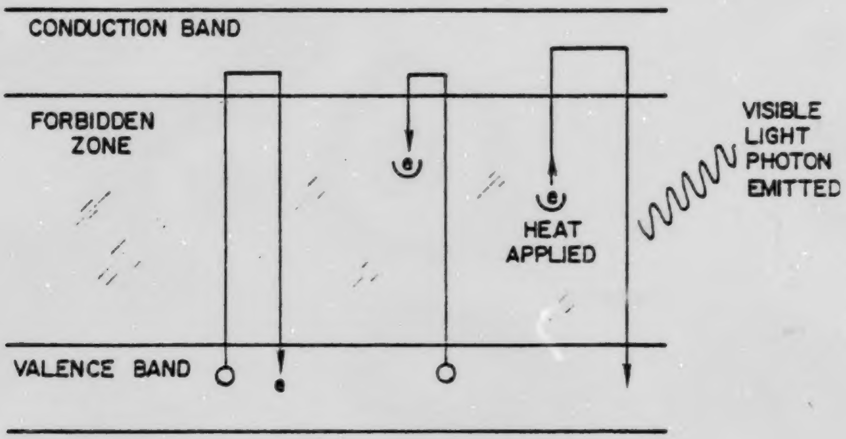


Figure 11-6-9. Solid-state diagram illustrating thermoluminescence. (Electron at left has dropped back into valence band immediately after being raised to conduction band by exposure to x- or gamma rays. Electron in middle has been trapped by imperfection in crystal and is at an energy level intermediate between valence and conduction band. Electron at right has been released from trap by application of heat - it falls back from conduction band to valence band with emission of a photon of light.)

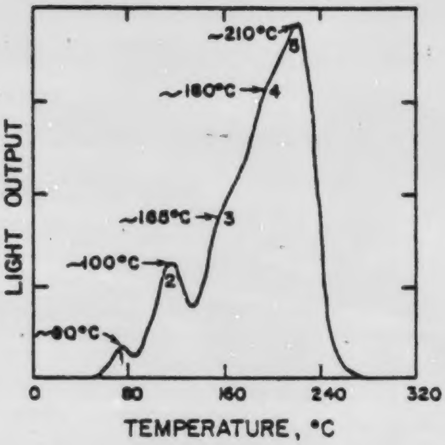


Figure 11-6-10.
Typical glow curve of LiF after annealing one hour at 400°C and irradiating 100 R. (Five peaks are shown.)

at approximately 80, 110, 155, 180 and 210°C. The location and height of the peaks are different in different crystals, and even in the same kinds of crystals if they have different impurities or have been annealed differently. (Annealing is a slow heating process that completely empties the traps and restores the crystal to its original state.) If properly annealed, a TLD can be reused.

The thermoluminescent response of a crystal is related to the impurities it contains. And the sensitivity of any given TLD can be adjusted by adding known and correct amounts of the proper impurities (known as "dopants"). Many different elements are used as dopants, with Mn, Mg and Dy commonly used. Common thermoluminescent materials are those that show desirable response to radiation and glow peaks at temperatures in the range 150 - 300°C. Glow peaks below about 150°C are unstable in that these so-called low temperatures tend to fade or release their electrons spontaneously at room temperatures. Indeed, the probability, P of a trap being emptied is a function of temperature as shown in Equation 11-6-5:

$$P = Se^{-(E/kT)} \qquad \text{Eq. 11-6-5}$$

where

P = probability of trap being emptied per unit time.
T = absolute temperature.
E = energy level of trap (trap depth).
k = Boltzmann's constant.
S = constant related to the atomic vibration frequency for the particular type of crystal.

If the temperature, T, is held constant, then only E and P are variables and, as E becomes small, as is the case with low-energy traps, P approaches 1. Temperatures higher than about 350°C are difficult to read out because the TLD is ordinarily heated on a metal pan that becomes incandescent at temperatures approaching 400°C. Other factors to be considered include wavelength of the light, light absorption with the crystal itself and effective atomic number of the TLD material.

Many TLD materials have been studied over the years and are currently used in personnel dosimetry and other applications. These include LiF, CaF_2, $CaSO_4$, $Li_2B_4O_7$, and BeO, usually doped with Mn. CaF_2 and $CaSO_4$ are particularly sensitive, and dysprosium doped CaF_2 is used for very low-level environmental dosimetry, being able to easily measure levels of 1 mR or less.

The most popular material is LiF, available in many forms, such as powder, rods or ribbons. The ribbons are tiny, measuring only 3.2 mm square by 0.5 mm thick; however, the effective range of a single chip of LiF, 3.2 x 3.2 x 0.5 mm, is from 10 mR (or less) to greater than 10^5 R.

This span of 7 orders of magnitude is at least a factor of 100 greater than can be achieved with RPL glasses and perhaps 10,000 times the span of a typical photographic film. TLD response or light output per unit exposure is essentially linear over a wide exposure range, but above about 1000 R, a phenomenon known as "supralinearity" is observed. Supralinearity means that, above a certain dose, the response per R tends to increase (Figure 11-6-11).

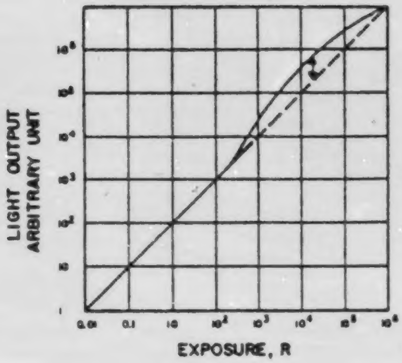


Figure 11-6-11.
Calibration curve for LiF. (This curve is linear to about 800 R and then becomes supralinear from that point. The linear relationship is shown as a dashed curve above 1000 R to allow comparison. At about 10^5 R, the curve begins to tail over - limit of the phosphor has been reached.

Perhaps the single most advantageous characteristic of the TLD, in addition to its wide dose response range, is its virtual lack of energy dependence, shown by the phosphors. LiF has an effective atomic number very close to that of tissue, providing a response to photons that is very close to that of tissue. It is, for all practical purposes, a dosimeter that responds very closely to the way tissue does, and is very nearly tissue equivalent.

LiB_4O_7 also has an effective Z approximating that of tissue or air, and BeO actually underresponds slightly to the lower energies since its effective Z is lower than either tissue or air.

The most widely used TLD, LiF, has certain other advantages. It can be enriched in ^7Li, which is essentially insensitive to neutrons, permitting the gamma or beta-plus-gamma dose to be accurately determined in a mixed field of neutrons, beta and gamma radiation. The broad range and neutron insensitivity of ^7LiF make this TLD material nearly ideal for measurements of gamma radiation in the core of a reactor. LiF can also be enriched in ^6Li and used as a thermal neutron dosimeter as well.

TLD's are hardly affected by environmental variables normally encountered in occupational situations - to a small extent by sunlight - and angular dependence is minimal because of the small size of the TLD's used. The TLD is read out by heating and measuring the light output with a photomultiplier reader system (Figure 11-6-12).

The TLD reader is usually designed to measure the total light output, but the height of a particular glow peak can also be measured. The TLD glow curve can be drawn and analyzed manually in addition to recording the reading of the instrument. Glow curves are sometimes retained as a legal record of the TLD reading.

At low dose levels, there may be some interference with the TLD reading from spurious or mechanically-induced thermoluminescence. This phenomenon is sometimes called "tribothermoluminescence" and can be eliminated by flowing an inert gas such as nitrogen or argon around the TLD as it is being heated. The gas quenches the spurious peaks, which are enhanced by the presence of oxygen.

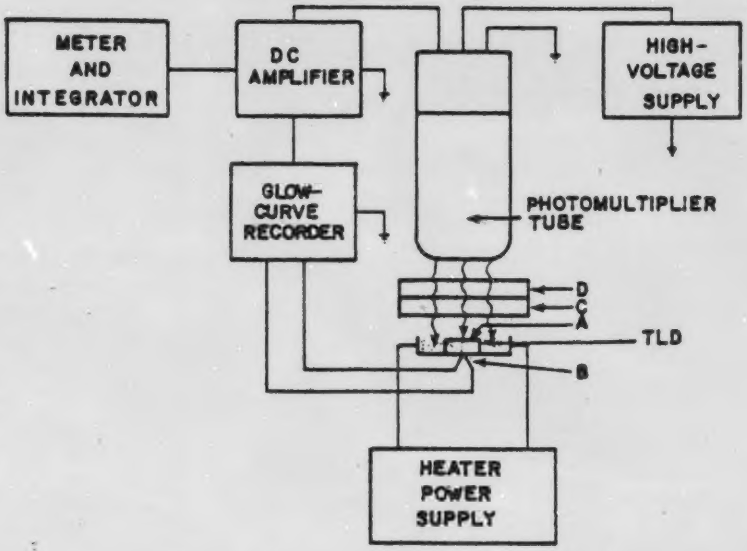

METER AND INTEGRATOR
DC AMPLIFIER
HIGH-VOLTAGE SUPPLY
GLOW-CURVE RECORDER
PHOTOMULTIPLIER TUBE
D
C
A
TLD
B
HEATER POWER SUPPLY

Figure 11-6-12. Schematic diagram of a thermoluminescent dosimeter reader. (A = heater pan containing thermoluminescent dosimeter; B = thermocouple in contact with the pan; C = an optical filter rejecting infrared light; D = a bandpass filter suitable for the thermoluminescent light.)

TLD's have been used for some interesting and unusual dosimetry applications. TLD techniques have been used on roof tiles in the Japanese cities of Hiroshima and Nagasaki to determine gamma-ray doses more than 20 years after the atomic bombs were detonated over those cities. The tiles were ground up into a powder and read out in a conventional manner. Gamma-ray exposures, down to a few hundred R, could be determined. The neutron sensitivity of these tiles was found to be small, permitting a reasonably accurate estimate of gamma-ray dose to be made. The data, comparable with other measurements and calculations where good comparisons can be made, are of value in determining and studying the radiological consequences of the detonations.

Age dating of rock can also be done with TLD techniques. By first determining the amount of uranium, potassium and other naturally occurring radioactive materials in the rock, dose rates of the rock can be calculated. A small amount of the rock can be ground up and read out by conventional TLD techniques. The dose is obtained by comparison with exposure of the same rock to a known dose. Once the dose to the rock has been determined, its age can then be simply calculated from the calculated dose rate.

As noted, the TLD can be reused since the readout empties the traps and returns them to their original state. But with the lower-temperature traps, not all of them are emptied, and the electrons in the high-temperature traps may drop down to the low-temperature traps. "Annealing" is then required to empty the traps and restore the TLD to its original state. Annealing is ordinarily accomplished by maintaining a short period of high temperature - well above the maximum thermoluminescence temperature - followed by a longer exposure period at a lower temperature, usually about 80°C. The specific times and temperatures are a function of the TLD material.

OTHER DOSIMETERS

The pocket ionization chamber (PIC), discussed in Module 11-2, "Gas-Filled Detectors I: Ionization Detectors," is a small integrating device widely used to provide a relatively prompt indication of exposure. It is seldom, if ever, used for long-term quantification of personnel exposure, primarily because of its inherent instability and lack of accuracy.

The PIC, generally quite energy dependent, tends to drift. In dusty or highly humid atmospheres, it may discharge and give a spuriously high reading. For these reasons, the PIC is usually used in conjunction with a photographic or luminescent dosimeter, which is considered to provide more accurate data.

In recent years, small electronic devices have become popular for individual use in radiation fields. Most of these are miniaturized GM instruments, but a few ion chambers are also used. There are three types of such instruments:

Ratemeters.
Dosimeters.
Combinations.

The ratemeters are usually designed to chirp or sound a constant tone of variable pitch when exposed to a given radiation field. Increasing the field will increase the frequency of the chirps or the pitch and intensity of the sound emitted. Some may be set to sound an alarm when a particular exposure rate has been reached.

The dosimeters are designed merely to act as integrating devices and, in a sense, miniature scalers. Ordinarily, they are equipped to sound an alarm when a preset exposure level has been reached.

The combinations combine the features of both the ratemeter and dosimeter type and are becoming more commonplace.

Obviously, these electronic instruments, commonly called "chirpers" or "personnel alarm dosimeters," are not dosimeters in the same sense as the personnel dosimeters described previously. As is the case with the PIC, these should be considered only as indicators of dose rather than true dose measuring devices.

EXERCISES

1. Contrast the response mechanism of radiophotoluminescent and thermoluminescent dosimeters.
2. Which type of personnel dosimeter would be best in a reactor containment building where temperatures to 50°C are expected and relative humidity averages 80%?
3. A developed film showed the following density pattern:

Filter	Density
Pb	0.13
Ag	0.24
Plastic	0.33
Open window	1.81

What can be said about the exposure? If the sensitivity to ^{60}Co gammas is 4.5, what is the dose from high-energy photons?

4. Use Equation 11-6-5; calculate probability of a trap being emptied in one minute if trap depth is equivalent to 80°C and $S = 10^2$.
5. Make a table that shows the number of rems/rad for α, β, γ, x-rays, thermal neutrons and fast neutrons.
6. Would a TLD phosphor with the composition $CaSO_4$ be energy dependent? Why? If the phosphor were doped with gold instead of manganese, would this make it

less energy dependent?

7. Why are personnel dosimeters not needed for alpha radiation?

MATERIALS

33 rolls commercial personnel dosimetry film (beta-gamma): 11 each of the three different speeds.
30 low Z filters.
1 densitometer.
1 TLD reader.
20 annealed TLD's (LiF is preferred).
1 radiation source.
3 TLD's with unknown exposure.
Graph paper (3 x 3 cycle log-log).
Plastic envelopes or coin envelopes.

PROCEDURES

PERSONNEL DOSIMETRY FILM

1. Prepare film for exposure as follows: Tape a low-Z filter (supplied by instructor) to one-half of each film.

2. Expose one film of each type to a source of mixed gamma and beta radiation for an exposure of 100 mR. Determine exposure with a GM survey meter.
3. Repeat No. 2 above for exposures of 200 mR up to 1 R.
4. Have exposed film developed. Also, have one unexposed control film of each type developed. (Students are not required to develop film.)
5. Measure densities of developed film (use densitometer); record densities on Data Table 11-6-1.
6. Subtract density of exposed films from the unexposed control film; record results (net densities).
7. Plot net density versus exposure for the three types of film, both with and without filters.
8. Compare curves with film manufacturer's data; explain any differences in Data Table 11-6-2.
9. Explain in data table when each of the three types of film would best be used.
10. Explain (in data table) effects of filters.

DETERMINATION OF TLD AND EXPOSURE
(If TLD Is Available)

1. Carefully handling TLD's, place them in plastic envelopes; expose two TLD's to each of the following levels: 0, 10, 20, 40, 100, 200, 400, 1000 and 4000 R.
2. After exposures, allow TLD's to set overnight, or for up to one week. (Store TLD's in dark place with low background.)
3. Read TLD's on reader, one at a time, recording read-

ing for each. (Internal light source should be used to check reader every 3 to 5 readings.)

4. Use data obtained to plot reading versus dose on log-log paper; make a calibration curve.
5. Read out three unknown TLD's and determine exposure to each; use calibration curve made in No. 4.
6. Verify results with instructor.

DATA TABLE 11-6-1

Film types:

#1: ______________________

#2: ______________________

#3: ______________________

Type filter: ______________________

Source type and activity: ______________________

			GROSS DENSITOMETER READINGS					
			Film #1		Film #2		Film #3	
Exposure	Survey Meter Reading	Exposure Time	with Fil	w/o Fil	with Fil	w/o Fil	with Fil	w/o Fil

DATA TABLE 11-6-2

NET DENSITOMETER READINGS

Control readings:

Film #1: ______________________

Film #2: ______________________

Film #3: ______________________

NET DENSITOMETER READINGS

	Film #1		Film #2		Film #3	
Exposure	with Filter	w/o Filter	with Filter	w/o Filter	with Filter	w/o Filter

Attix, F.H. Luminescence Dosimetry. U.S. Atomic Energy Commission Symposium Series, No. 8. Springfield, VA: U.S. Atomic Energy Commission, Office of Information, Technical Information Center, 1967.

________, ed. Topics in Radiation Dosimetry. New York: Academic Press, 1972.

________ and Roesch, W.C., eds. Radiation Dosimetry, Vol. 2. New York: Academic Press, 1970.

Becker, K. Photographic Film Dosimetry. New York: Focal Press, 1966.

________. "Photographic, Glass or Thermoluminescence Dosimetry." Health Physics 12(1966):955.

________. "Recent Progress in Radiophotoluminescence Dosimetry." Health Physics 14(1968):17.

Cameron, J.R.: Suntharalingam, N.; and Kenney, G.N. Thermoluminescence Dosimetry. Madison, WI: University of Wisconsin Press, 1968.

Francois, E.; Gupton, E.D.; Maushart, R.; Piesch, E.; Somasundaram, S.; and Spurny, Z. Personnel Dosimetry Systems for External Radiation Exposures. International Atomic Energy Agency Technical Reports Series, No. 109. Vienna: International Atomic Energy Agency (IAEA), 1970.

International Commission on Radiological Units and Measurements (ICRU). "Radiation Quantities and Units," Report No. 11. Washington, DC: National Bureau of Standards, 1971.

Kathren, R.L. "Plant and Personnel Radiation Monitoring." in Nuclear Power Reactor Instrumentation Systems Handbook, Vol. 2, Ch. 13. Edited by J.M. Harrer and J.G.

Beckerley. Springfield, VA: U.S. Atomic Energy Commission, Office of Information, Technical Information Center, 1974.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. Define the following terms:

   a. Roentgen: ______________________________

   ____

   b. Rad: ______________________________

   ____

   c. Rem: ______________________________

   ____

   d. Quality Factor: ______________________________

   ____

   e. Radiophotoluminescence: ______________________________

   ____

   ____

   f. Gray: ______________________________

   ____

2. Why is it incorrect to refer to a person's dose in terms of Roentgens?

   ____

   ____

3. For each unit of radiation listed below, determine whether the unit applies to exposure, absorbed dose, activity, quality factor, or radiation protection. Fill in the correct application for each unit.

Unit	Application
a. Roentgen	______________
b. Rem	______________

c. Seivert

d. Curie

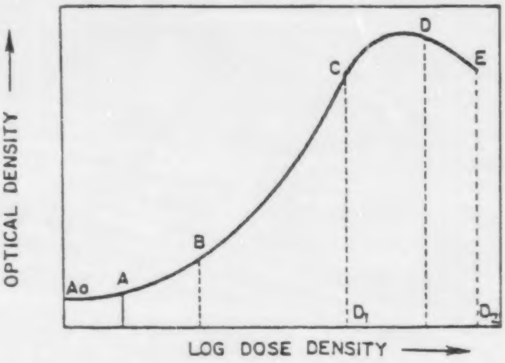
e. Linear Energy Transfer

4. For what types of radiation are personnel dosimeters used? Why?

5. Describe the response of a photographic film dosimeter with silver-activated phosphate glass to ionizing radiation. Be sure to include the purpose of the latent image in your response.

6. What is meant by the term "optical density" in a photographic film dosimeter? How does radiation affect optical density?

7. Study the characteristic response curve of a photographic film exposed to ionizing radiation and answer the questions below it.



a. In the A_o to A region, how is density related to dose? Is the film usable for dosimetry in the A_o to A region?

b. In the region from A to B, how is density related to dose? Is the film usable for dosimetry in this region?

c. In what region is the film the most accurate?

8. What does the term "sensitivity" of the film mean?

9. What methods are commonly used to increase the sensitivity of film dosimeters?

10. What causes energy dependence in the use of film dosimeters? How can this energy dependence be minimized?

11. How does the radiophotoluminescent dosimeter react when exposed to ionizing radiation?

12. What is the composition of typical radiophotoluminescent glass?

13. How does the use of ultraviolet light after irradiation affect the radiophotoluminescent dosimeter?

14. Why is RPL (radiophotoluminescent) glass as a dosimeter superior to photographic film for radiation detection?

15. What are three drawbacks to the use of RPL glass?

    a. ______________________________

    b. ______________________________

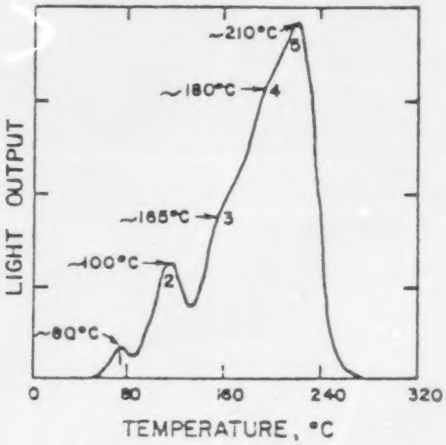
    c. ______________________________

16. Briefly describe what happens to the electrons of the thermoluminescent dosimeter's (TLD) inorganic crystal when exposed to radiation.

17. What is the purpose of the dopant?

18. Why is the dosimeter's crystal exposed to heat after it has been exposed to radiation?

19. Here is a typical glow curve for LIF used in a TLD. Use the curve to answer parts a, b, and c of this question.



a. What does the glow curve show?

b. What causes the glow peaks?

c. Would lithium fluoride (LIF) be a good inorganic crystal to use in a TLD? Why?

20. Name two TLD materials, besides lithium fluoride.

a.

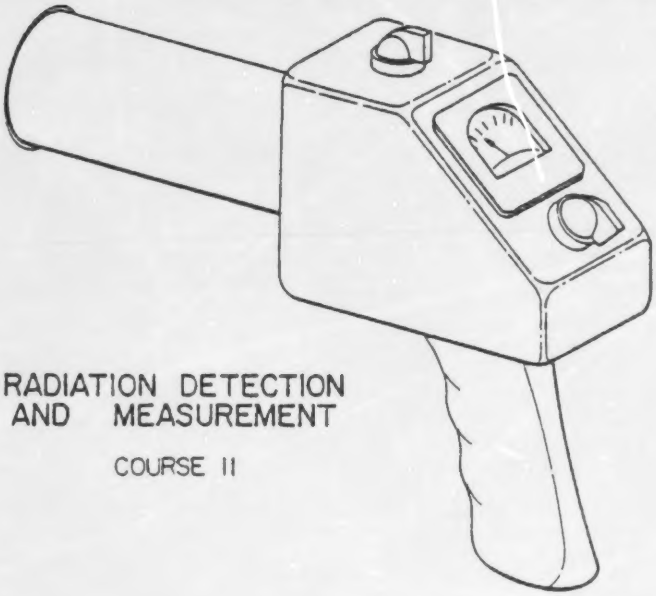
b.

21. What type of dosimeter, photographic emulsion, RPL or TLD, would you select given a broad band spectrum of gamma radiation? Why?

22. Compare photographic film, RPL's and TLD's used for personnel dosimetry as to the following characteristics: Extent of environment, range and energy dependence.

NUCLEAR TECHNOLOGY

MODULE 11-7 NEUTRON DETECTORS, DOSIMETRY AND ACTIVATION ANALYSIS.



RADIATION DETECTION AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
⇨ MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

For more than 30 years, scientists were concerned only with detection and measurement of charged particles (α and β rays) and electromagnetic radiation in the form of gamma and x-rays. But, in 1930, two German scientists, Walther Bothe and Hans Becker, reported that a highly penetrating form of radiation was obtained when very light elements (beryllium, boron and lithium) were bombarded with alpha particles from the radioisotope ^{218}Po. In 1932, the French husband and wife team, Frederic and Marie Joliot-Curie observed that, when a layer of hydrogenous material was placed in the path of this highly penetrating radiation, protons were ejected.

Later that year, Sir James Chadwick showed that this highly penetrating radiation was, in fact, a heretofore undiscovered particle with no charge and a mass of 1 amu, the neutron.

For his discovery, Chadwick was awarded the Nobel Prize in physics for 1935. The Joliot-Curies, whose explanation of the proton recoils they observed was in error, and who barely missed the discovery themselves, were awarded the Nobel Prize in chemistry in the same year for their discovery of artificial radioactivity, which grew out of their experiments with the strange new radiation that Chadwick identified.

The neutron is an uncharged particle, and therefore its mechanism of interaction with matter differs from that of charged particles or electromagnetic radiation. Detection of neutrons requires interactions with matter to obtain charged particles, these being measured and related back to the neutron flux. Neutron interactions, however, are highly energy dependent - much more so than gamma radiation interactions with matter.

Recall that the cross section for neutron interactions with many nuclei follows a 1/2 (inverse) proportionality to neutron energy. Thus a 1-MeV neutron might have a cross section or probability of interaction that is only 0.001 the cross section for a 1 keV neutron, and less than 10^{-6} the probability of an interaction with a thermal neutron.

This proportionality, however, is only a part of the problem, for many nuclei exhibit "resonances," or peaks in cross section for neutrons of a particular energy.

Determining neutron dose is more difficult than detecting neutrons, or even measuring their numbers. The dose process is complicated by the fact that it is the energy of the secondary particles (produced by the neutron interactions) that produce the dose. And, if it is the dose equivalent or biological dose that is to be measured, the problem becomes even more complex, because the quality factor of the neutron is highly dependent on its energy.

Despite all the difficulties and problems, it is possible to detect and measure neutrons with a reasonable degree of accuracy. The dosimetry of neutrons can be accomplished in reasonable fashion, although the accuracy is not nearly as great as can be achieved for beta and gamma radiation.

Several of the earlier modules in this series have touched upon neutron detection and measurement. All of the detectors discussed in Modules 11-2 through 11-6 can be adapted to the detection of neutrons. But, neutrons have certain special and unique properties that make their detection, measurement and dosimetry far different from that of charged particles and x or gamma rays.

In this module, the techniques of neutron detection are reviewed, and the principles and methods of personnel neutron dosimetry are presented. And, a wholly new topic - neutron activation analysis - is introduced, including a discussion of its applications to neutron measurement, as well as other areas.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe three general groupings of neutron detection methods and list applications for each.
2. Describe basic principles and applications of a compensated ionization chamber.
3. Given the appropriate equipment or data, calculate current produced by an ionization detector used as a mean-level neutron detector.
4. List three common activation foils.

5. Describe how nuclear emulsions can be used for monitoring neutrons, and for neutron dosimetry.
6. Explain the operation of an albedo neutron dosimeter.
7. Explain the energy dependence of neutron dose; draw a curve of dose equivalence quality factor as a function of neutron energy.
8. Describe how a fission-fragment dosimeter works.
9. Explain neutron activation analysis and its applications and advantages.

DETECTION OF NEUTRONS

Because neutrons are uncharged, they do not interact to any great extent with electrons, and produce no significant primary ionization as they travel through matter. Unless neutron fluxes are extremely large, detection by means of primary ionization is not possible. So, neutron detection and measurement methods are based on secondary effects.

For convenience, neutron detection methods can be placed into three groupings, depending on the secondary effects produced:

Instantaneous detection of charged particles produced by neutron-induced transmutation.

Delayed detection or measurement of radioactive species produced by neutron interactions.

Elastic scattering of neutrons in which recoil nucleus is charged and, therefore, capable of detection.

The first of these groupings is primarily used for detection of slow neutrons, including such reactions as (n,α), (n,γ), (n,fission) and (n,p) in which charged particles of photon are ejected essentially instantaneously.

Such reactions are useful for determining neutron rates; the most important reaction is the $^{10}B(n,\alpha)^{7}Li$ with slow neutrons. This (and other similar reactions that produce charged particles) was discussed in the section on neutron proportional counters in Module 11-2, "Gas-Filled Detectors I: Ionization Detectors."

MEAN-LEVEL AND PULSE-COUNTING SYSTEMS

It should be noted that although neutron reactions releasing charged particles were discussed in the context of proportional counters, the same reactions can also be used with ionization chambers. There are applications for BF_3-filled ion chambers, boron-lined ion chambers and fission-ionization chambers. But a proportional counter is generally chosen because it gives a larger, more easily detected pulse, and because gamma discrimination is more easily accomplished.

In many instances, the gamma field that almost always accompanies a neutron flux may be large enough to interfere with neutron measurements. Thus, a neutron counter may not provide accurate readings in a high-level $n+\gamma$ field (remember, neutrons are almost always accompanied by gamma radiation).

In a pulse-counting system, a pulse-height selector (discriminator) is used to prevent smaller pulses produced by the gamma rays from entering or triggering the counting circuit. When gamma pulse pile-up is sufficient to produce a pulse big enough to pass the discriminator, the effective upper range of the counter has been reached (the detector is no longer able to resolve individual pulses).

In mean-level detectors (ionization chambers), the problem of gamma interference is present, even at low levels, because the instrument measures current irrespective of what type of radiation produces the current.

Gammas and neutrons can be separated by means of "compensated ionization chambers," which allow monitoring of neutrons in mixed fields. In its purest form, a compensated ion chamber has two chambers with identical responses to gamma radiation. But one chamber is lined with boron, or contains a fissionable material, so that it will respond to neutrons as well. Thus, one chamber responds to gammas only, and the other responds to both neutrons and gammas. The currents produced by gamma rays in the two chambers electronically offset each other, or balance one another (Figure 11-7-1), and the instrument reads out only the current produced by the neutrons.

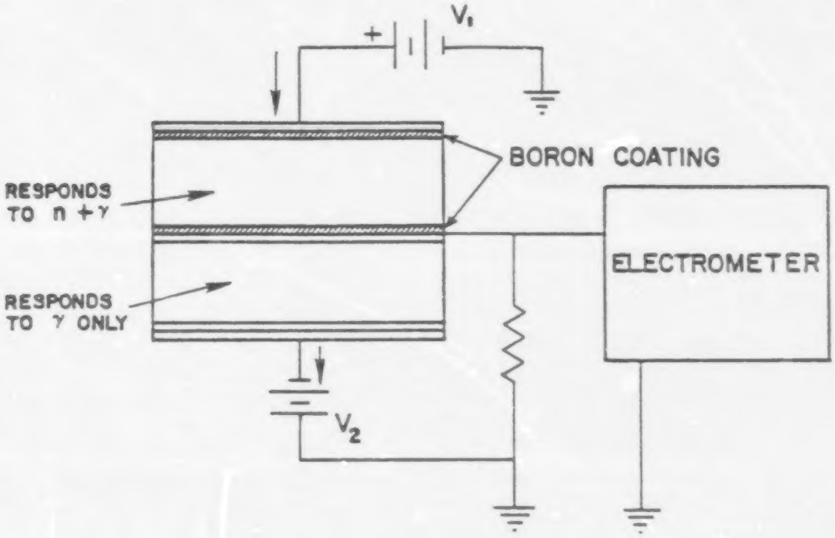


Figure 11-7-1. Compensated ionization chamber, with two chambers using a common center wall.

The response or current of any ionization chamber used as a mean-level counter for detection of neutrons can be calculated:

$$I = \frac{kVN\sigma\phi E}{W}$$

Equation 11-7-1
(Equation = Eq.)

where

I = current, in amps.

k = constant equal to 6.28×10^{18} charges per second per amp.

V = volume of the detector, in cm^3.

N = number of atoms (e.g., ^{10}B) available for neutron interactions per cm^3 of the detector.

σ = cross section (or chance that reaction will occur in cm^2).
(Note: Cross section is discussed in Course 12, "Reactor Physics.")

ϕ = neutron flux, in n/cm^2-sec.

E = energy released by charged particles produced by nuclear reaction, in eV.

W = energy expended to produce an ion pair, in eV.

Equation 11-7-1 assumes a detector filled with neutron-sensitive material (gas). If walls are boron lined, for example, the terms V and N refer to the boron liner; therefore, a factor must be applied to take into account the fraction of the charged particles that will escape the lines and actually get into the sensitive volume of the chamber; also, those that do will have only a fraction (ranging from 0 to 100%) of the kinetic energy available to them, since they may have lost some of that energy in escaping from the wall.

In addition to gas-filled detectors (ionization chambers and proportional counters), neutron rate measurements can also be made with scintillators. There are many scintillators that can be used for this purpose: plastic scintillators, rich in hydrogen; proton recoils, produced in crystal (directly counted with a photomultiplier and standard counting system. Note: The sensitivity of such scintillators to gamma interactions in the crystal can be electronically discriminated out.)

The most common neutron scintillator is LiI(Eu), which counts alpha particles produced in the scintillator by a $^{5}Li(n,\alpha)^{3}H$ reaction. The associated detector has a relatively low sensitivity to gamma radiation, which, because of its smaller pulse size, is easily discriminated out. Pulse pile-up is minimal in scintillators, generally, because of the small resolving time.

ACTIVATION DETECTORS

The second grouping of neutron detectors are those that rely on delayed (after-the-fact) measurement of radioactive products produced by neutron interactions. This procedure involves measuring the radioactivity induced in a material and then calculating the neutron flux. (Details on how to make this calculation are presented in a later module.)

RESONANCE FOILS

Other portions of the neutron energy spectrum can be measured by "resonance foils" or "threshold foils." A resonance foil is one that has a high activation probability, or peak, at a specific energy or group of energies; the cross section under this peak is known as the "resonance integral." If properly selected, the nuclide will have a resonance integral so large that virtually all activation will be produced by neutrons in the energy region covered by the resonance.

Resonance integrals for ^{115}In, which is commonly used as a foil for measuring neutron flux, are shown in Figure 11-7-2.

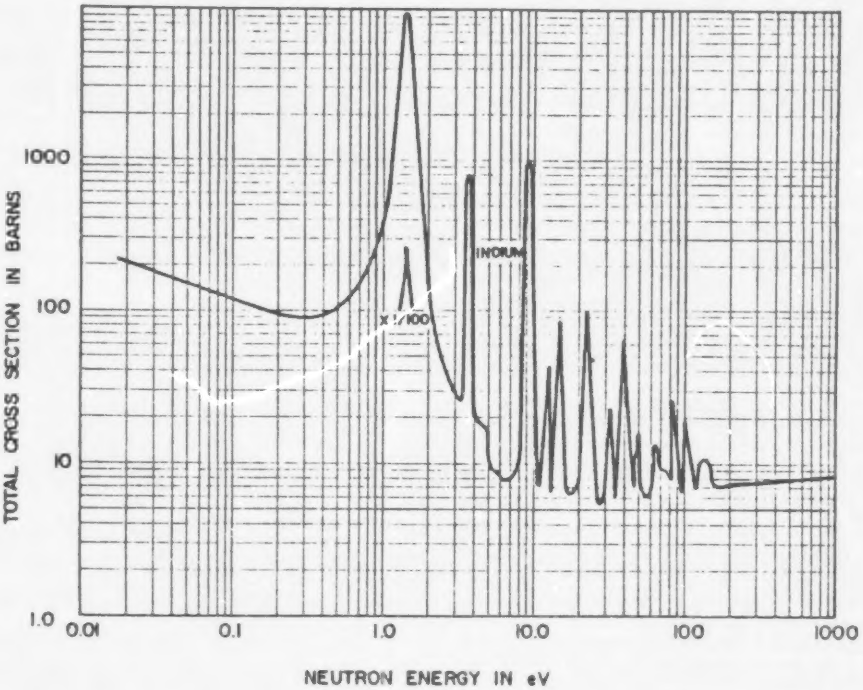


Figure 11-7-2. Cross section of ^{115}In. (Note large number of resonance peaks between 1 and 100 eV.)

Another method of neutron activation involves the use of threshold foils, which are made from nuclides with a definite cutoff (or energy level) below which they do not respond to neutrons. Table 11-7-1 shows a range of common-

TABLE 11-7-1

PROPERTIES OF COMMONLY USED THRESHOLD FOILS

REACTION	ISOTOPIC ABUNDANCE	NEUTRON THRESHOLD ENERGY (MeV)	MEAN CROSS-SECTION FOR FISSION NEUTRONS (BARNS)	HALF-LIFE OF PRODUCT
$^{24}Mg(n,p)^{24}Na$	78.6%	4.9	1.3	14.96 Hr.
$^{27}Al(n,\alpha)^{24}Na$	100%	3.3	0.63	14.96 Hr.
$^{32}S(n,p)^{32}P$	100%	1.0	61.1	14.28 Days
$^{56}Fe(n,p)^{56}Mn$	91.7%	2.9	0.96	2.6 Hr.
$^{115}In(n,n')^{115m}In$	95.8%	0.35	164	4.5 Hr.
$^{237}Np(n,f)$		1	1355	
$^{238}U(n,f)$		1.5	310	

ly used threshold foils and their neutron threshold energies and cross sections. Note that the last two foils listed make use of the fast fission of ^{237}Np and ^{238}U; any radioactive fission product of known yield and convenient half-life can be counted to determine neutron flux above threshold energy.

Neutron activation foils have the following characteristics and advantages:

Passive and do not have maintenance problems or limited shelf life.

Small in size.

Insensitive to gamma radiation.

Controlled sensitivity over a wide range, accomplished by appropriate foil selection and foil size adjustment.

Induced β or γ activity is conveniently and easily measured with laboratory counting equipment after irradiation (and outside neutron field).

Indication of neutron energy range by appropriate selection of foils.

It should be stressed that activation foils are "integrating devices," and give "average flux" over the period of exposure; they are not ratemeters. Thus, activation foils are useful for measuring "fluence" or n/cm^2 (nvt) and for neutron dosimetry.

NUCLEAR EMULSIONS

The last group of neutron detectors to be discussed uses elastic scattering interactions between neutrons and hydrogen nuclei. This group is useful for detecting high-energy neutrons - that is, neutrons with energies above about 1 MeV.

In the elastic scattering reaction, a neutron collides with a hydrogen nucleus, dislodging it and imparting some energy to it, producing a moving hydrogen nucleus or proton recoil.

Proton-recoil detectors can be either ratemeters or integrating types; ratemeters can be ion chambers or proportional counters filled with hydrogenous gas or lined with hydrogenous material.

Scintillation detectors can also be made to respond to neutrons if made primarily from hydrogenous material.

The most common proton-recoil detectors, however, are integrating types: photographic emulsions. Over the years, these photographic materials, known as "nuclear emulsions," have been successfully used for detection and quantification of neutrons, and for determining neutron energy spectrums.

Nuclear emulsions are small, passive devices, inexpensive and easily handled. They are subject to all the environmental difficulties associated with β-γ film dosimetry, however.

A photographic film designed for neutron detection has a fairly thick emulsion, usually around 30 μm, on a base about 200 μm thick (Figure 11-7-3).

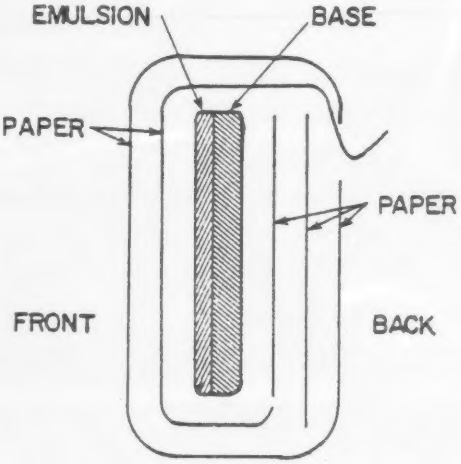


Figure 11-7-3.
Nuclear emulsion film packet for neutron monitoring.

The film is wrapped in paper, which is also hydrogenous to give rise to protons that may enter the film emulsion and be detected. Actually, protons may arise from several sites and be detected in the sensitive portion of the film. These sites include the emulsion itself, the film base, added hydrogenous materials and the wrappers (Figure 11-7-4).

As neutrons travel through the film, proton recoils leave behind electrons that, in turn, produce low-energy secondary electrons. These secondary electrons produce

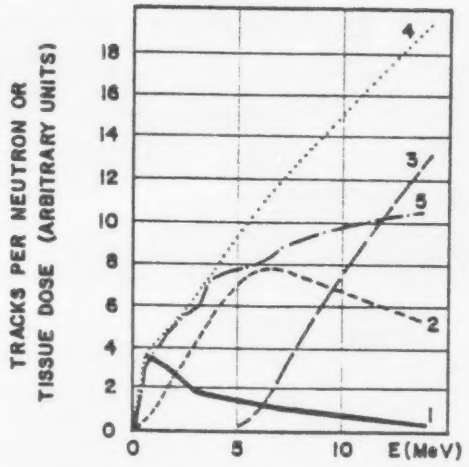


Figure 11-7-4.
Response of a nuclear emulsion film packet as a function of neutron energy. Note: Curve 1 is response from emulsion; Curve 2 is response from film base; curve 3 is response from 227 mg/cm² of paper (hydrogenous material) in front of film; Curve 4 is total of Curves 1-3; Curve 5 is dose equivalent curve.

latent images in the grains along the path that each proton recoil has traveled. When the film is developed, a track of developed grains is left behind (Figure 11-7-5).

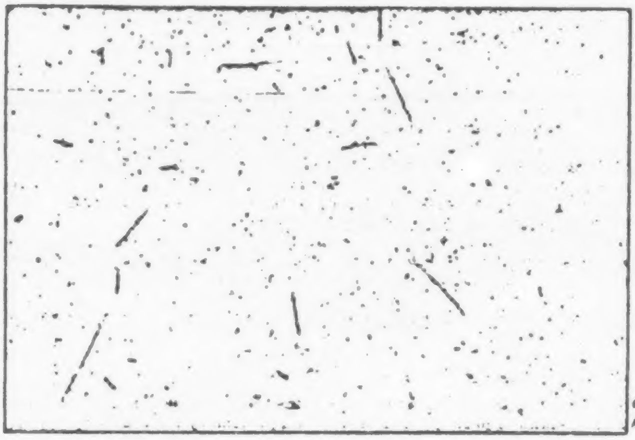


Figure 11-7-5.
Proton recoil tracks produced in nuclear track emulsion.

Energy of a recoil proton is proportional to energy of the incident neutron, according to the relationship shown in Equation 11-7-2:

$$E_p = E_n \cos^2\Theta \qquad \text{Eq. 11-7-2}$$

where

E_p = energy of proton.

E_n = energy of incident neutron.

Θ = angle between incident neutron and recoil proton.

Because track length is proportional to energy (or range) of the proton, it is also proportional to the energy of the neutron. Hence, over a limited range, nuclear emulsions can be used to determine energy of an incident neutron. [This range is about 450 keV to 20 MeV. Below 450 keV, the neutron is unable to impart sufficient energy to the proton to form a trace, and above 20 MeV, neutrons produce "stars" or clusters of tracks by inelastic collisions, causing breakup of the heavier nuclei (Ag, Br, S, C, etc.) in the emulsion.]

Proton recoil tracks are quite small, but can be visually counted through a microscope, usually at a magnification of 400X to 1000X, which provides a field size about 200 µm in diameter (or 0.03 mm in area).

Because there may only be a few tracks in each field, many fields must be counted to obtain good statistics, often 100 or more. Automatic scanners available for nuclear emulsions automatically count tracks over a large area of the film.

As was noted earlier, nuclear emulsions are subject to the same environmental effects as are photographic films; however, nuclear emulsions are particularly susceptible to latent image fading, especially in high humidities. And, gamma exposure blackens the film, making it impossible to distinguish proton recoil tracks. The energy range of nuclear emulsions can be broadened by adding materials that produce other charged particles from interactions with lower-energy neutrons. Thus, emulsions are sometimes loaded with β or other nuclides that have high cross sections for the (n,α) reaction, or with nitrogen to take advantage of the $^{14}N(n,p)^{14}C$ reaction. This reaction has a thermal cross section of 1.76 barns and produces protons with an energy of 0.63 MeV, sufficiently long to be counted by ordinary methods.

Ordinary photographic films can be used with activation to measure thermal neutrons. A material, such as cadmium (Cd) or indium, with a high thermal neutron cross section, is placed next to a wrapped film. When exposed to neutrons, the Cd is activated, and the beta particles pass through the film wrapper, causing the film to blacken. The shorter the half-life, the better; a long half-life material takes a long time to irradiate the film.

PERSONNEL NEUTRON DOSIMETRY

Neutron dosimetry is made difficult because neutron effects are highly energy dependent. Neutrons of different energies deposit different amounts of energy in tissue; that is, the quality factor (Table 11-7-2) varies with neutron energy (Figure 11-7-6).

TABLE 11-7-2

THE INTERRELATIONSHIPS OF NEUTRON ENERGY, QUALITY FACTOR, FLUX AND DOSE

Neutron Energy (MeV)	Quality Factor	Flux (In n/cm^2-sec) per mrem/hr	Fluence(n/cm^2) per rem x 10^{-8}
2.5×10^{-8} (thermal)	2.3	260	9.5
1×10^{-7}	2	240	8.7
1×10^{-6}	2	220	8.0
1×10^{-5}	2	230	8.3
1×10^{-4}	2	240	8.6
1×10^{-3}	2	270	9.8
1×10^{-2}	2	280	10.2
1×10^{-1}	7.4	48	1.75
5×10^{-1}	11	14	0.52
1	10.6	8.5	0.31
2	9.3	7.0	0.25
5	7.8	6.8	0.24
10	6.8	6.8	0.24
20	6.0	6.5	0.23
50	5.0	6.1	0.22
1×10^{2}	4.4	5.6	0.20
2×10^{2}	3.8	5.1	0.18
5×10^{2}	3.2	3.6	0.13
1×10^{3}	2.8	2.2	0.079
2×10^{3}	2.6	1.6	0.058

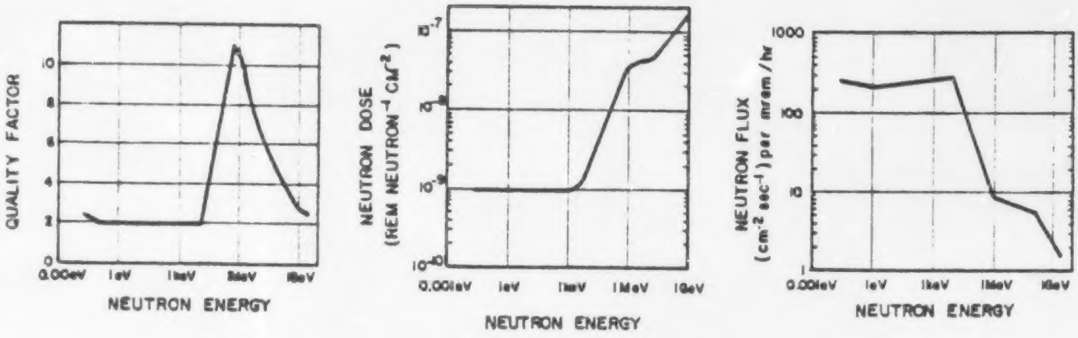


Figure 11-7-6.

Quality factor as a function of energy.

Neutron flux equivalent rate of 1 mrem/hr as a function of energy.

Neutron dose equivalent to human body per unit fluence as a function of neutron energy.

Neutrons, being uncharged, produce their dose as a result of secondary particles, including protons, heavy recoil nuclei, and gamma rays. For convenience, neutrons are put into four broad energy groups (Table 11-7-3).

TABLE 11-7-3

NEUTRON ENERGY GROUPINGS

GROUP	NERGY RANGE	QUALITY FACTOR
Slow (thermal)	≤ 0.5 eV	3
Epithermal (or intermediate)	0.5 to < 10 keV	5
Fast	10 keV to 10 MeV	10
Relativistic	> 10 MeV	7.5

As a practical matter, relativistic neutrons are seldom encountered and, in most neutron spectra, fast neutrons are the major contributions to dose. As a spectrum becomes richer in low energies, these fast neutrons account for a larger fraction of the dose, even though the dose per neutron becomes smaller.

PERSONNEL NEUTRON DOSIMETERS

Because the neutron spectrum is generally unknown, personnel neutron dosimeters must either respond to neutrons of all energies - the way the human body does - or separate the neutrons into energy groups from which appropriate dose contributions can be calculated. The earliest, and still probably the most common, neutron dosimeters used photographic films. Nuclear emulsions, which were used to measure the fast-neutron component from about 0.5 to 10 MeV, also provided, in some cases, the slow-neutron dose from tracks produced by the $^{14}N(n,p)$ reaction. Such dosimeters missed the intermediate-neutron energy contribution entirely. But, because most of the dose comes from fast neutrons, this drawback may not be too serious.

The intermediate-energy neutron spectrum could be better indicated by placing a sheet of cadmium over a personnel dosimetry film for β-γ monitoring and subtracting out the blackening caused by gamma dose. (Cadmium passes fast neutrons but absorbs the slow ones.) The remaining density could be attributed to slow (and some intermediate) neutrons; dependent upon gamma dose, neutron doses as low as 1 mrem could be detected. With nuclear emulsions, the lower limit of detection is usually around 100 mrem. (A skilled technician, counting 100 or more fields, might be able to detect doses from fast neutrons as low as 20 mrem.)

Activation foil, overlying an integrating device, can be used with RPL glass and TLD as well as ordinary film; however, RPL glass and TLD lend themselves better to neutron dosimetry in other ways. For example, RPL glass or mica can be used as a fission-fragment dosimeter. In this kind of a dosimeter, a fission foil is placed against the glass or mica. Fission fragments produced in the foil penetrate into the glass or mica and produce tiny damage centers. These centers are amplified (etched) by soaking the glass in an alkaline or acid solution to produce microscopic pits. These pits can then be counted under magnification and correlated with the number of fissions in the foil to derive neutron fluence. (Neutron fluence is integrated neutron flux over time.)

Fission-fragment neutron dosimetry is limited by the energy response of the fissionable material used, with a usable range from a few mrem up to about 100 rem. Table 11-7-5 gives detection limits and energy ranges for a five-foil dosimeter. A major drawback to fission-fragment neutron dosimetry is its excessive cost, and the need to release, in relatively uncontrolled fashion, quantities of fissile and radioactive materials.

In the early 1970's, albedo neutron dosimeters using TLD's began to be used for personnel monitoring. The neutron dosimeter, in its simplest form, uses two TLD's - one wrapped in a material such as cadmium or indium, which readily absorbs slow but not fast neutrons, and the other wrapped with an equivalent thickness of a material that does not

TABLE 11-7-5
FISSION-FRAGMENT DOSIMETER CHARACTERISTICS

Fission Foil	Neutron Energy	Lower Detection Limit (mrem)	Comments
^{235}U	Slow	10^{-3}	Foil is Cd-covered to eliminate thermal neutron response
^{235}U	1 MeV	1	
^{238}U	2-6 MeV	1	
^{232}Th	3-6 MeV	0.5	
$^{237}N_p$	1-5 MeV	5	

readily absorb neutrons. Thus, both TLD's show the same response to gamma radiation. If the TLD is ^{7}LiF, it has no direct neutron response, and only the TLD wrapped in cadmium or indium shows neutron response.

The response comes from activation of the TLD wrapped in cadmium or indium, which is made into a short-lived radionuclide by exposure to neutrons. These dosimeters are known as albedo neutron dosimeters because they are worn on the body, and the neutrons are reflected by the body back into the dosimeter. (Scattered or reflected neutrons are known as albedo neutrons, hence the name of the dosimeter.)

A great advantage of the albedo neutron dosimeter is its ability to detect neutrons in the range of 5 eV to about 200 keV, where most other dosimeters do not detect.

Although the albedo neutron dosimeter can detect neutrons in this intermediate energy range, it is highly energy dependent and tends to overrespond to low-energy neutrons. This overresponse can be corrected to a great extent by a a calibration factor if the spectrum is reasonably well known. But the same statement could be made, in general, for many other types of dosimeters. Note also that the albedo dosimeter (Figure 11-7-7) must be worn close to the body to work properly.

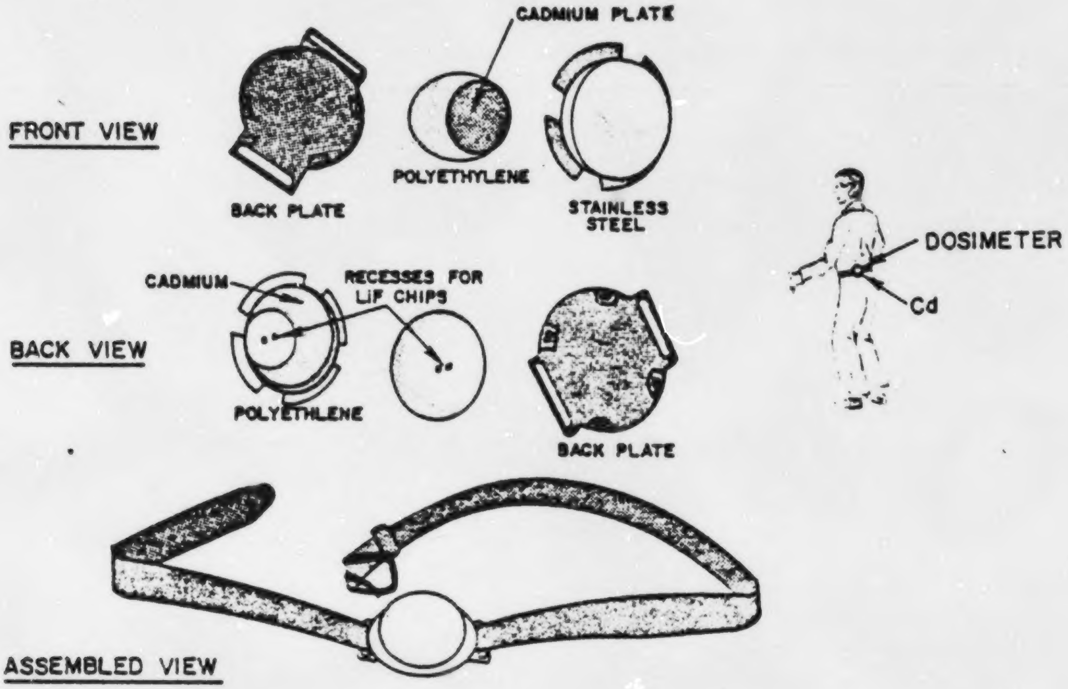

CADMIUM PLATE
FRONT VIEW
POLYETHYLENE
BACK PLATE
STAINLESS STEEL
DOSIMETER
Cd
CADMIUM
RECESSES FOR LiF CHIPS
BACK VIEW
POLYETHLENE
BACK PLATE
ASSEMBLED VIEW

Figure 11-7-7. Thermoluminescent albedo-neutron dosimeter badge. Note: The badge fits onto belt back and is sensitive only to neutrons passing through the body, or fast neutrons from behind being reflected from body into badge. Slow neutrons from behind body are absorbed by Cd plate. (Adapted from Hoy)

The albedo concept can be used with film or RPL glass, as well as TLD, although (as shown in the previous module)

TLD is a superior dosimeter.

Thermal neutron dosimetry can be relatively easily accomplished with LiF TLD's. If enriched in ^{6}Li, a dosimeter has great response to slow neutrons; response to 1 rem of slow neutrons is approximately 80 times response to 1 R of gamma radiation. Thermoluminescent materials can be used to monitor slow neutrons in high-level gamma fields. Also, gamma doses in mixed n-γ fields can easily be determined by ^{7}LiF TLD.

Neutron activation of various elements is perhaps the most widely used method for high-level or accident dosimetry or for fixed area monitoring. The human body, and clothing normally worn, can provide measurable activity following neutron irradiation of sufficient magnitude. Gold rings, watches and other jewelry can be removed and counted following high-level exposure to neutrons to obtain data regarding thermal neutron exposure. Similarly, silver objects can be removed and analyzed for radioactivity, and their activation correlated with fast neutron fluence.

Two elements in the body are particularly useful for activation measurements because they readily absorb neutrons and are widely distributed in the body. Hair contains an appreciable amount of ^{32}S that is transmuted to ^{32}P by an (n,p) reaction. This ^{32}P, which is a pure beta emitter with a half-life of 14.3 days, can be chemically extracted from hair or nails and counted directly.

The gamma activity of ^{24}Na, produced by a ^{23}Na (n,γ) reaction, can be measured from a sample of blood withdrawn for that purpose or by placing the exposed individual in a whole-body counter.

A whole-body counter is a device for measuring gamma ray-activity of the human body. Essentially, it consists of a shielded room in which subject is placed with one or more very large and highly sensitive NaI(Tl) crystals that feed into a multi-channel analyzer. In addition to measuring activation of body sodium, whole-body counters (also known as "in-vivo" counters) can be used to measure gamma-emitting radionuclides in the body (from natural or occupational exposures).

Many activation foils, having been incorporated into personnel dosimeters, are worn to provide neutron dose estimates in event of a nuclear criticality accident. A typical dosimeter may contain several foils, designed to cover a wide range of neutron energies. One such dosimeter (Figure 11-7-8) uses a combination of indium, cadmium-covered indium, gold, cadmium-covered gold, sulfur and cadmium-covered copper.

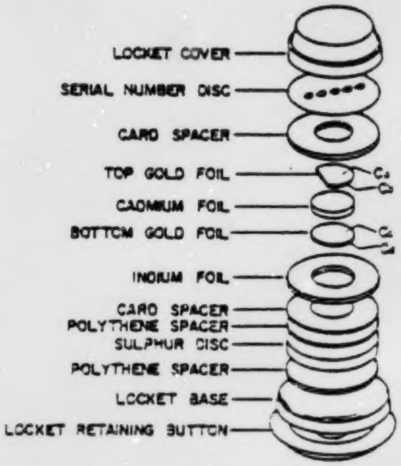

LOCKET COVER
SERIAL NUMBER DISC
CARD SPACER
TOP GOLD FOIL
CADMIUM FOIL
BOTTOM GOLD FOIL
INDIUM FOIL
CARD SPACER
POLYTHENE SPACER
SULPHUR DISC
POLYTHENE SPACER
LOCKET BASE
LOCKET RETAINING BUTTON

Figure 11-7-8.
Exploded view of neutron dosimeter with several activation foils. By counting each foil after exposure and combining information, a series of mathematical calculations can be made to break the neutron spectrum into six energy intervals. (Note: Particular neutron dosimeter shown is used at Harwell, England, by UK Atomic Energy Authority for personnel monitoring.)

Thus, a reasonably accurate assessment of neutron dose can be made, although the intermediate energy range is not well broken down. A thermoluminescent or radiophotoluminescent glass dosimeter can be included along with the foils to provide high-level gamma dose capability.

NEUTRON ACTIVATION ANALYSIS

Neutron activation analysis (NAA) is a process whereby unknown substances are bombarded with neutrons and then analyzed by gamma spectroscopy to determine the various elements present. Amounts present may be quantified by simple calculations.

For many elements, neutron activation analysis permits a level of sensitivity that cannot be achieved with other analytical methods. For example, gravimetric (by weight) analysis allows for detection in the parts per thousand region; colorimetric, spectrographic and mass spectroscopy are usually limited to parts per million. Neutron activation analysis, however, is frequently useful in the parts per billion range. More than fifty elements can be detected in amounts of 1 nanogram (10^{-9} g) or less.

Neutron irradiation can be accomplished with reactors, accelerators or by neutron sources. Research reactors are capable of producing slow fluxes of 10^{12} to 10^{13} n/cm^2-sec, with some going as high as 10^{15} n/cm^2-sec. Accelerators can be used to produce fast neutrons. A small, compact unit, relatively low in price, is available to produce 14-MeV neutrons from a $^3H(d,n)^4He$ reaction.

Practical applications of NAA have been many, ranging from trace element analyses in human tissues and body fluids to analyses of moon rocks. A fairly common application is in police work, where material lifted from the hands of a suspect can be analyzed to ascertain whether or not suspect has recently fired a gun. If so, material taken from the hand that held the gun will contain minute amounts of metals (found in gunpowder) that were driven into the hand when the gun was fired. (Each particular type of ammunition has a characteristic pattern; i.e., each contains slightly different ratios of trace elements - particularly antimony, barium and copper. Thus, it also may be possible to identify the type of ammunition used.)

EXERCISES

1. Calculate current produced when an ion chamber with a volume of 100 cm^3, lined on its inner walls with 1 g of boron, is exposed to 4 x 10^7 n/cm^2-sec. Assume that cross section value is 945 x 10^{-24} cm^2, and W = 34.7 eV/ion pair. (Use Equation 11-7-1.) Would this ion chamber be as sensitive as one filled with BF_3?
2. A technician, working around a Van de Graaff accelerator producing 14-MeV neutrons, may have been exposed. How could you determine dose if technician was not wearing a dosimeter?
3. The great physicist, R. Gone, was wearing a neutron foil dosimeter that included a bare indium foil and one wrapped in cadmium. Counting rate of the bare dosimeter was 11,400 dis/min, and the wrapped one was 2,000 dis/min. What can you tell about the spectrum?

1 neutron source (^{238}PuBe).
1 portable survey meter with external bare BF$_3$ detector, or bare BF$_3$ detector connected to scaler.
6 paraffin or polyethylene sleeves with wall thicknesses from 1 to 15 cm, designed to fit around BF$_3$ tube.
Graph paper.
Cadmium sheet or sleeves approximately 5 mm thick.

PROCEDURES

NEUTRON DETECTOR RESPONSE AS A FUNCTION OF NEUTRON ENERGY

1. Under direction of the instructor, set up neutron source and make reading with bare detector; record data on Data Table 11-7-1.
2. Repeat procedure with each paraffin (or polyethylene) sleeve around tube.
3. When all readings have been taken and recorded, plot count rate as a function of sleeve thickness; use graph paper. Which sleeve thickness gave the maximum reading? Why?
4. Determine sensitivity of BF$_3$ detector; use equation:

$$S = \frac{\text{cpm}}{\text{n/cm}^2\text{-sec}} .$$

5. Repeat experiment with a cadmium sheet wrapped around outside of paraffin sleeve. What effect did cadmium have outside the paraffin? Why?
6. Repeat experiment with cadmium inside paraffin sleeve. What effect did it have inside the paraffin? Why?

DATA TABLE 11-7-1

Neutron Source (Type and Strength): ____________________

Distance from Source to Detector Tube: ________________

Instrument Used: __________________________________

PART A		PART B		PART C	
No Cadmium		Cadmium Outside Sleeve		Cadmium Inside Sleeve	
Sleeve Thickness	Count Rate	Sleeve Thickness	Count Rate	Sleeve Thickness	Count Rate

REFERENCE MATERIALS

Attix, F.H. and Roesch, W.C., eds. *Radiation Dosimetry*, Vols. 1 & 2. New York: Academic Press, 1970.

Corliss, William R. "Neutron Activation Analysis." Understanding the Atom Series. Springfield, VA: U.S. Atomic Energy Commission, Office of Information, Technical Information Center.

Francois, E.; Gupton, E.D.; Maushart, R.; Piesch, E.; Somsaundaram, S. and Spuruy, Z. *Personnel Dosimetry for External Radiation Exposures*. International Atomic Energy Agency Technical Reports Series, No. 109. Vienna: International Atomic Energy Agency (IAEA), 1970.

Hoy, Jack E. "An Albedo Type Personnel Dosimeter." *Health Physics* 23 (1972):385.

Instrumentation for Environmental Monitoring - Radiation, LBL-1. 2nd update. 3 (October 1973).

International Atomic Energy Agency (IAEA). *Neutron Monitoring*. Proceedings of IAEA Symposium. Vienna: International Atomic Energy Agency, 1967.

International Commission on Radiological Units and Measurements (ICRU). "Radiation Protection Instrumentation and Its Applications," Report 20. Washington, DC: National Bureau of Standards, 1971.

Price, W.J. *Nuclear Radiation Detection*. New York: McGraw-Hill, 1965.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. Why must neutron detection and measurement be based on secondary effects?

2. Neutron detection methods can be placed in three groupings, depending on the secondary effects produced. Name and very briefly describe the three general groupings.

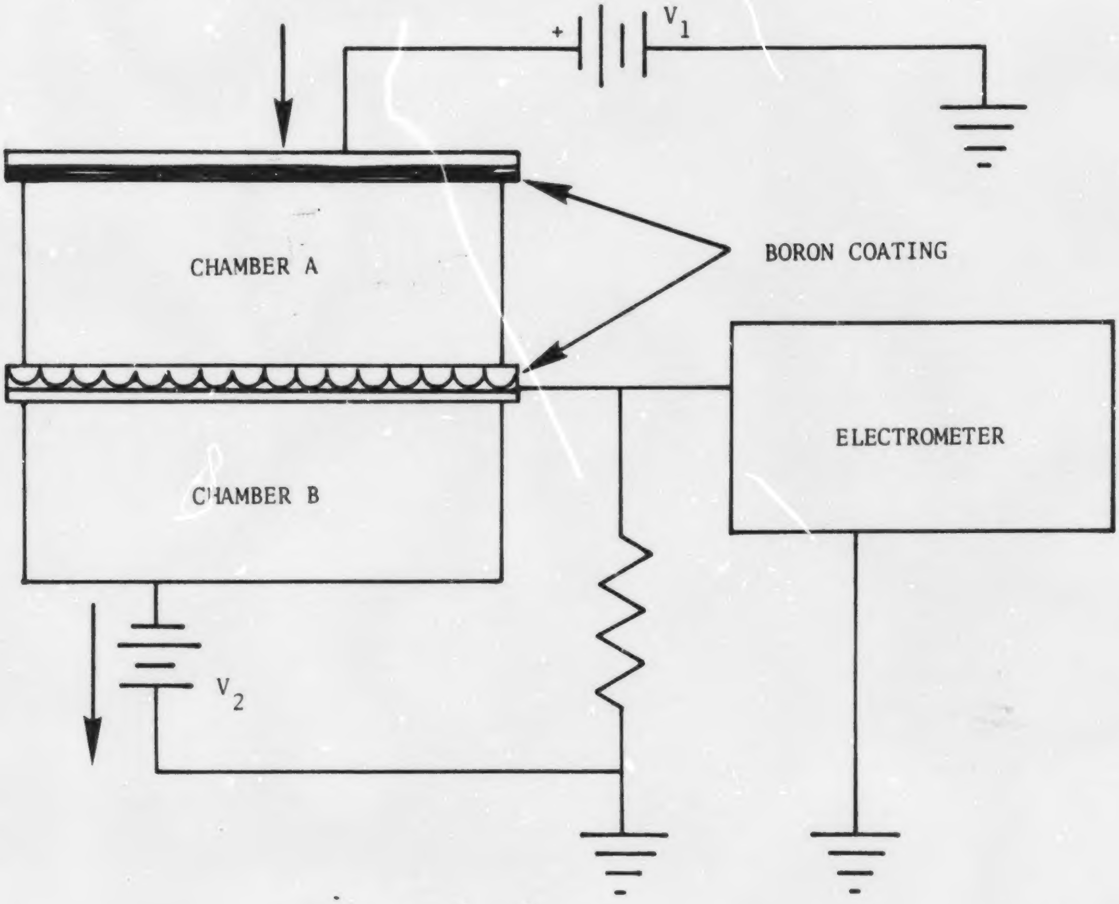
   a.

   b.

   c.

3. Why can't a mean-level detector with an uncompensated ion chamber be used to detect neutron radiation?

4. Below is a drawing of a compensated ionization chamber. Use the drawing to answer questions 4a.-4e.


+
V_1
BORON COATING
CHAMBER A
ELECTROMETER
CHAMBER B
V_2

a. Why does a detector need the capability to perform neutron monitoring in a mixed field?

b. What is the purpose of the Boron Coating?

c. To what type(s) of radiation will Chamber A respond?

d. To what type(s) of radiation will Chamber B respond? ____________

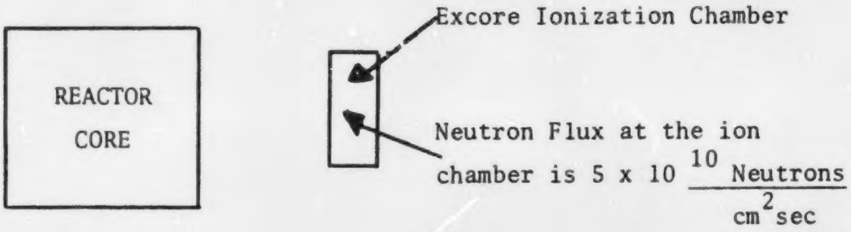
____

e. What does the detector read out? Why? ____________

____

____

5. A pressurized water reactor is operating at 100% power. This reactor employs an excore ionization chamber for monitoring the neutron flux of the core as shown in the figure below:



Neutron Flux at the ion chamber is $5 \times 10^{10} \frac{\text{Neutrons}}{\text{cm}^2\text{sec}}$

If the neutron flux at the detector location is $5 \times 10^{10} \frac{\text{neutrons}}{\text{cm}^2\text{sec}}$, calculate the output current, in amperes, developed by the ion chamber assuming the following:

$K = \dfrac{1 \text{ ampere}}{G.28 \times 10^{18} \frac{\text{Charges}}{\text{Sec}}}$

$V = 200 \text{ cm}^3$ (Detector Volume)

$N = 7.43 \times 10^{18} \dfrac{{}^{10}\text{Batoms}}{\text{cm}^3}$ (^{10}Batom density in the detector)

$\sigma = 3.838 \times 10^{-21} \dfrac{\text{cm}^2}{{}^{10}\text{Batoms}}$ (Microscopic cross section for ^{10}Batom)

$E = 2.31 \times 10^{6} \dfrac{\text{eV}}{\text{Neutron}}$ (Energy released in the detector for each neutron event.)

$W = 35 \dfrac{\text{eV}}{\text{Charge}}$ (Energy required to produce a unit of electron charge in the ion chamber.)

Make your calculations on the next page.

6. List two common activation foils:

   a. ____________________

   b. ____________________

7. Answer these questions about neutron detectors that make use of the elastic scattering reaction.

   a. What is proton recoil? What causes proton recoil? ____________________

   ____

   ____

   b. What are nuclear emulsions? For what purposes are they used? ____________________

   ____

   ____

   ____

   c. In films designed for neutron detection, what do the grains on the developed film represent? ____________________

   ____

d. What relationship does grain track length have to the energy of the neutron?

8. Neutron dosimetry is difficult because neutron effects are highly energy dependent. What does "highly energy dependent" mean as applied to neutron radiation?

9. Use the grid below and the chart of neutron energy groupings, to draw a curve showing quality factor as a function of energy in neutron detection.

Neutron Energy Grouping	
Group	Energy Range
Slow	$\leq$ 0.5 ev
Epithermal	0.5 to < 10 Kev
Fast	10 Kev to 10 Mev
Relativistic	> 10 Mev

10. Answer these questions about fission fragment neutron dosimeters.

a. In a fission fragment neutron dosimeter, why is a fission foil placed against the RPL glass or mica?

b. How are the pits that are etched in the RPL glass by soaking used?

11. Answer these questions about albedo dosimeters.

a. In what neutron energy range is the albedo detector sensitive?

b. Why must the albedo detector be worn close to the body, i.e., back of the belt?

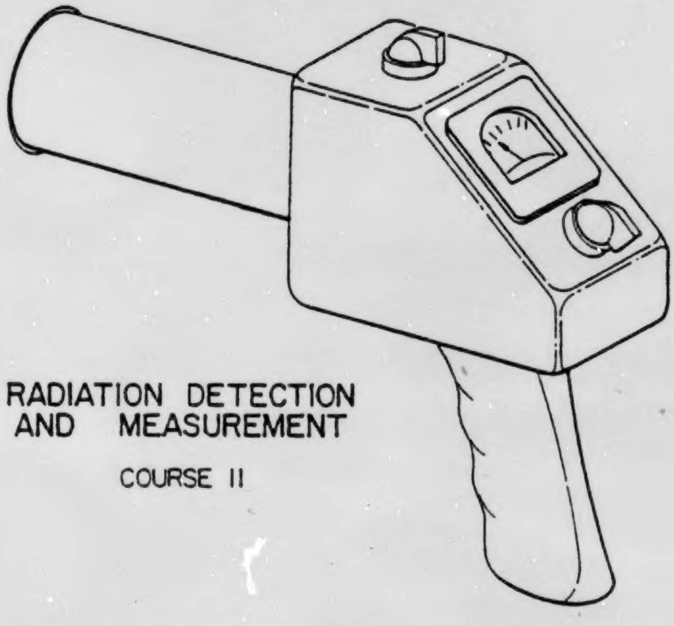
c. Describe how the albedo dosimeter's thermoluminescent devices work to measure exposure.

12. What is a whole-body counter? What does it measure?

13. Briefly explain neutron activation analysis.

NUCLEAR TECHNOLOGY

MODULE 11-8 RADIOLOGICAL CALIBRATION AND STANDARDIZATION.



RADIATION DETECTION
AND MEASUREMENT

COURSE II

COURSE MODULES

COURSE 11 RADIATION DETECTION AND MEASUREMENT

MODULE 11-1	Statistics of Radiation Counting
MODULE 11-2	Gas-Filled Detectors I: Ionization Detectors
MODULE 11-3	Gas-Filled Detectors II: Proportional Counters
MODULE 11-4	Gas-Filled Detectors III: Geiger-Mueller Counters
MODULE 11-5	Scintillators, Semiconductors and Spectroscopy
MODULE 11-6	External Personnel Dosimeters for Beta-Photon Monitoring
MODULE 11-7	Neutron Detectors, Dosimetry and Activation Analysis
⇨ MODULE 11-8	Radiological Calibration and Standardization

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INTRODUCTION

In the previous seven modules of this course, radiation detection and measurement were discussed, along with applications. Only brief mention was made of how good these systems are for measuring radiation and what yardsticks are used to determine usefulness.

In day-to-day use, the words "accuracy" and "precision" are often used interchangeably (and erroneously). In science and engineering, these words have specific meanings.

Accuracy refers to how closely an instrument measures a true or correct value. For example, if an object weighs 150 pounds, but the scale records only 135 pounds, an error of 15 pounds (10% of true value) occurs.

While accuracy refers to deviation from the true or correct value, precision refers to the repeatability or reproducibility of a measured or observed value. Thus, if a person weighed 100 times on the same scale and always weighed exactly 135 pounds, there would be perfect precision. But, if that person weighed 10 times and got the following values - 135, 138, 137, 140, 141, 134, 139, 138, 142, 135 - the mean would be 138 pounds, which is imprecise.

Precision and accuracy are related in that precision can affect the uncertainty of any individual reading or measurement; accuracy, however, has almost no effect on precision.

Often, a statement of accuracy or uncertainty includes precision. Unfortunately, no standard way of expressing accuracy that is adhered to consistently exists, despite efforts and definitions provided by such standards-setting bodies as the National Bureau of Standards, International Union of Pure and Applied Physics, and International Commission on Radiation Units and Measurements.

MODULE PREREQUISITES

The student should have completed a course in basic algebra and Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe basic principles of radiological and radioactivity standards.
2. Differentiate between precision and accuracy.
3. Describe four methods of absolute radioactivity standardization.
4. Briefly explain what is meant by traceability.
5. Describe how a 4π counter works and why it is useful for radioactivity standardization.
6. Describe two commonly used methods of neutron source standardization.
7. Explain what is meant by the Bragg-Gray principle (cavity ionization method).
8. Define Γ.
9. Explain what is meant by electronic equilibrium.
10. Describe what a certificate of standardization should contain.
11. Differentiate between reference, standard and mock-standard sources.

12. Prepare a standard alpha source and determine its absolute activity by using a 2π ionization counter.
13. Calibrate a GM survey meter by using both a laboratory standard (known source) and a Condenser-R meter.
14. Calculate source strength when flux is given.

RADIOACTIVITY STANDARDIZATION

"Radioactivity" standardization refers to laboratory procedures and measurements designed to determine directly the radioactive disintegration rate of a radionuclide. A distinction is made between radioactivity and "radiological" standardization, which refers to measurements of exposure or dose, often by indirect methods, as will be discussed in the next section. In either case, standardization (calibration) procedures involve comparisons of unknowns with known standards.

A standard that has been calibrated in accepted units of mass, length and time (meter, kilogram and second, as in the Systeme Internationale [SI]; or centimeter, gram and second, as in the metric system) is known as an "absolute" standard.

Various national and international absolute standards are kept. In the United States, the National Bureau of Standards serves as the keeper and publisher of all U.S. national absolute standards (also known as "primary" standards). Standards that are calibrated against these (or are traceable to them) are known as "secondary" standards.

Absolute standardization of a radioactive source can be accomplished in many ways, but most commonly the following methods are used:

Measurement of disintegration rate.
Mass determination (weighing).
Calorimetry.
Ionization measurement.

MEASUREMENT OF DISINTEGRATION RATE

Most commonly, disintegration rate is measured by a pulse-counter type instrument. Detection efficiency of such an instrument must be well known for this method to be used. In the laboratory, calibration (i.e., determining detection efficiency) of counting equipment is accomplished by measuring a source of known strength. The instrument can then be used to standardize other sources.

Thus, the calibration and standardization procedure is a two-way operation: standard sources can be used to calibrate (or standardize) instruments, and instruments can be used to calibrate (or standardize) sources. Ultimately, all sources and instruments are traceable to one known primary (absolute) standard, which is kept by the National Bureau of Standards and has been calibrated against the NBS primary standards of mass, length and time (Figure 11-8-1).

Counting or determination of disintegration by direct means is accomplished in many different ways. For alpha and low-energy beta emitters, standardization is frequently done by 4π counting. A 4π counter is one in which the source can be placed inside the chamber so that all particles emitted by the source are counted. Thus, for alphas and betas, efficiency of the 4π counter is 100%. The source itself must be small enough so that self-absorption is virtually zero. Also, the source must be mounted on a thin film to avoid backscattering and absorption within the mounting.

The 4π counter derives its name from the fact that it subtends a solid angle of 4π steradians with the source. Ionization chambers, proportional counters and Geiger-

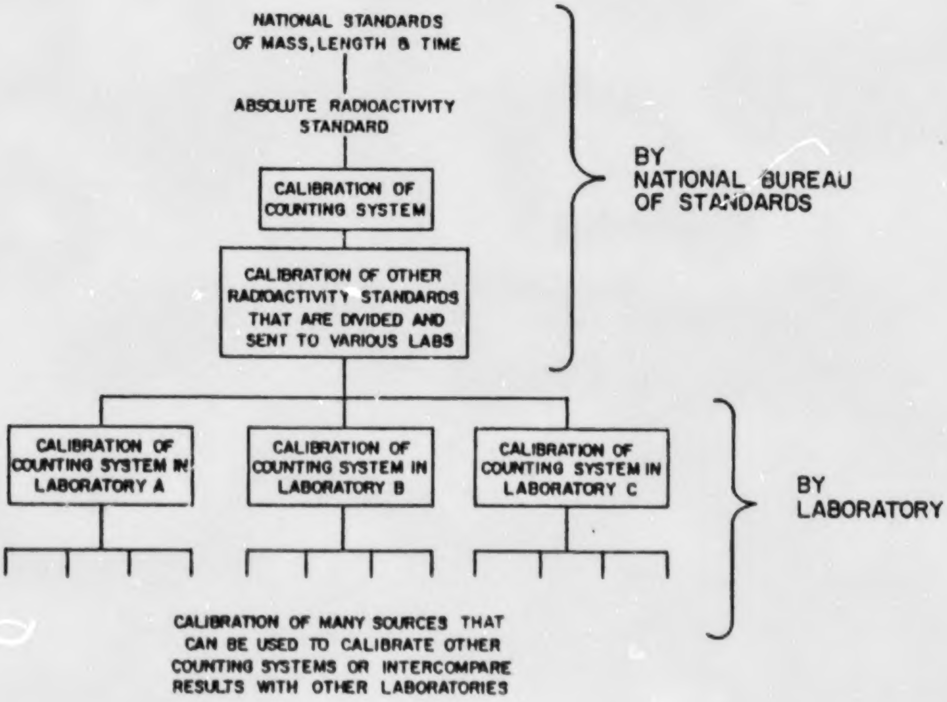
NATIONAL STANDARDS OF MASS, LENGTH & TIME
ABSOLUTE RADIOACTIVITY STANDARD
CALIBRATION OF COUNTING SYSTEM
CALIBRATION OF OTHER RADIOACTIVITY STANDARDS THAT ARE DIVIDED AND SENT TO VARIOUS LABS
BY NATIONAL BUREAU OF STANDARDS
CALIBRATION OF COUNTING SYSTEM IN LABORATORY A
CALIBRATION OF COUNTING SYSTEM IN LABORATORY B
CALIBRATION OF COUNTING SYSTEM IN LABORATORY C
BY LABORATORY
CALIBRATION OF MANY SOURCES THAT CAN BE USED TO CALIBRATE OTHER COUNTING SYSTEMS OR INTERCOMPARE RESULTS WITH OTHER LABORATORIES

Figure 11-8-1. Flow path for traceability by National Bureau of Standards; how sources and systems can be calibrated.

Mueller counters have been used as 4π counters, but the proportional counter shows certain advantages, including a shorter resolving time and a pulse larger than the GM pulse. Compare the 4π beta counter (Figure 11-8-2) with the proportional counter shown in Figure 11-3-5.

Alpha particles can also be standardized by 2π counting - essentially, the equivalent of counting in half of a 4π system. Alpha particles, because they are more massive than betas, are not as readily scattered as betas.

Other methods involving direct determination of radioactive disintegrations include coincidence counting and

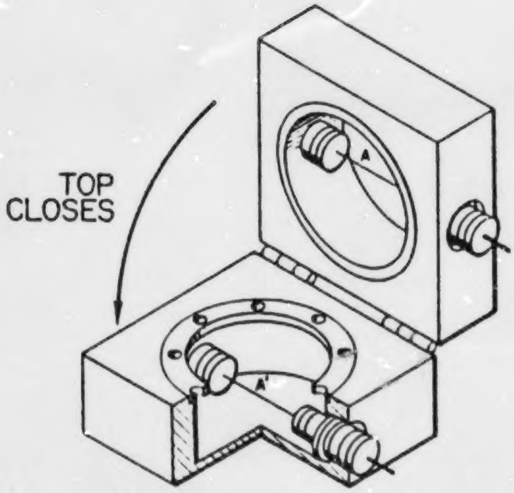

Figure 11-8-2.
Pillbox type 4π counter. "Weightless" sample is suspended in center; collecting electrodes (A, A') are located above and below sample.

scintillators. Solutions of radioactivity can be standardized by liquid scintillation techniques or by well counting methods. Liquid scintillation eliminates the problem of self-absorption, but requires great care in removing and preparing an aliquot of the sample to avoid introduction of errors.

MASS DETERMINATION

Another method of absolute calibrating - weighing, or mass determination - though seldom used is highly accurate. Mass determination is also the oldest form of standardization, having been used by Madam Curie in 1911 to obtain the first international standard of activity. This first standard was taken as being equal to the disintegration rate of exactly 1 gram of radium in equilibrium with its daughter products, and was the origin of the "curie"

(Ci) as the unit of radioactivity. (Note: The modern definition of the curie is 3.7×10^{10} nuclear transformations per second. But the curie is not the unit of activity in the SI system of units adopted by science internationally. The SI unit is the Becquerel [abbreviated Bq], which is equal to 1 disintegration per second; however, because the curie is a much more convenient unit and has been so widely used for more than half a century, the international body agreed that it could continue to be used for a time. One Ci = 3.7×10^{10} Bq.)

If weighing is to be used for absolute standardization, the purity of the nuclide must be known, as must its decay constant. The disintegration rate, $\frac{\Delta N}{\Delta t}$, can then be calculated by the form of the fundamental decay law:

$$\frac{\Delta N}{\Delta t} = \lambda N$$

Equation 11-8-1
(Equation = Eq.)

where

λ = decay constant in reciprocal seconds
= $\frac{0.693}{T_{\frac{1}{2}}}$ (where $T_{\frac{1}{2}}$ is the half-life in sec).

N = number of atoms of nuclide, as determined from weighing.

ΔN = number of atoms disintegrating in time interval, Δt.

$$N = \frac{\text{Avogadro's number}}{\text{atomic weight}} \times \text{weight in grams;}$$

$$\text{thus, } \frac{\Delta N}{\Delta t} = \lambda \times \frac{\text{Avogadro's number}}{\text{atomic weight}} \times \text{weight in grams.}$$

Weighing is a useful means of standardization for nuclides with low specific activities; i.e., where activity (in curies) per gram is very small, as with nuclides with very long half-lives.

EXAMPLE

Given: 100 grams of ^{238}U.
Atomic weight = 238.0508 amu.
Half-life = 4.51×10^9 years.

Find: Activity, A.
Number of atoms, N.

Solution:

$$\begin{aligned} A &= \frac{\Delta N}{\Delta t} = \lambda \times \frac{\text{Avogadro's number}}{\text{atomic weight}} \times \text{weight (grams)} \\ &= \frac{0.693}{4.51 \times 10^9 \text{ yrs}} \times \frac{6.023 \times 10^{23} \text{ atoms}}{238.0508 \times 90 \text{ amu grams}} \times 100 \text{ grams} \\ &= 0.389 \times 10^{14} \text{ dis/year} \\ &= 1.23 \times 10^6 \text{ dps} \\ &= 33.2 \text{ microcuries.} \end{aligned}$$

$$\begin{aligned} N &= \frac{\text{Avogadro's number}}{\text{atomic weight}} \times \text{weight} \\ &= \frac{6.023 \times 10^{23} \text{ atoms} \times 100 \text{ grams}}{238.0508 \text{ grams}} \\ &= 2.53 \times 10^{23} \text{ atoms.} \end{aligned}$$

CALORIMETRY

Calorimetry is a measurement of heat associated with radioactive decay, and is particularly suited to short-range particles such as alphas and low-energy betas. If the average energy per disintegration is known, activity can be calculated:

$$\text{Calories} = \frac{53.7\ E}{m} \qquad \text{Eq. 11-8-2}$$

where

E = average energy per disintegration, MeV.
m = mass of absorbing medium (calorimeter), grams.

IONIZATION MEASUREMENT

If the average energy to form an ion pair in a gas is known, ionization measurements can be used to accurately determine activity from a source. If an ionization chamber is used as a mean-level current-measuring device to accomplish this measurement, activity can be calculated from current, I:

$$I = \frac{(\text{dps})\ E}{k\ W} \qquad \text{Eq. 11-8-3}$$

where

I = current in amperes.

k = constant equal to number of ions per second per ampere
= 6.28×10^{18}.
dps = number of radioactive disintegrations per second (number of Bq).
E = average energy per disintegration, eV.
W = energy required to form an ion pair, usually taken to be 34 eV.

Note that, in ionization measurements, not only does average disintegration energy need to be known, but also the energy required to form an ion pair. If these values are in error, the error is reproduced in the standardization result or calculation.

SOURCE PREPARATION

A very important consideration in any radioactivity standardization is preparation and handling of the source. The source may be in the form of a liquid or solid or, less commonly, a gas.

Solid sources for radioactive standardization are usually electroplated onto thin metal discs, or deposited by evaporation onto a disc from a known volume of a standardized solution.

The source must be carefully handled to avoid loss of sample or disturbance of source. With sources made by the evaporation method, a thin coating of an organic resin may be sprayed or applied to keep the source intact and in place. This thin coating is satisfactory for beta emitters, but will adversely affect alpha emitters. For this reason, alpha emitters are commonly electroplated, which provides an adherent, controllably thin, uniform layer of the nuclide, and requires no cover. Among the disadvantages of electroplating are cost and the fact that not all materials can be conveniently electroplated onto thin metal discs.

Radioactive standard solutions are usually provided in sealed glass containers. Such solutions should be chemically and physically stable. Because an alpha emitter generates He gas, consideration must be given to the buildup of gas pressure within a container. Quantities of stable isotopes of the radionuclide may be added to minimize "plate out" or adsorption onto container walls, but appropriate chemical compound and solvent should be chosen. Table 11-8-1 provides examples of the composition of solutions

TABLE 11-8-1

COMPOSITION OF TYPICAL RADIONUCLIDE SOLUTIONS

Nuclide	Compound	Solvent
^{32}P	H_3PO_4	$\leq$ 0.5 N HCl
^{35}S	H_2SO_4	0.1 N HCl
^{35}S	Elemental S	Benzene (C_6H_6)
^{82}Br	KBr	1 N HCl
^{99}Tc	NH_4TcO_4	1 N NH_4OH
^{131}I	NaI	$NaSO_3 + NaOH$
^{198}Au	$AuCl_3(HAuCl_4)$	1 N HCl + HNO_3
^{210}Bi	$Bi(NO_3)_3$	1 N HNO_3

of common radionuclides. Note that most of the solutions are acidic; acid solutions reduce deposition or adsorption onto ampoule walls.

STANDARDIZED SOLUTIONS

Standardized solutions fall into two categories: absolute or "absolute standardized" solutions and "reference" solutions. Absolutely standardized solutions are those for which the disintegration rate has been accurately determined by direct methods without reference or comparison to other sources or standards of activity. Thus, one of the four methods described above would be used along with expert interpretation and calculation of results to obtain a highly accurate (uncertainty less that ± 1%) determination of activity.

Because absolute standardized solutions are expensive, they are frequently prepared in fairly large quantities from which many aliquots can be removed. Aliquots must be removed carefully, using appropriately calibrated volumetric equipment, such as micropipettes or pycnometers.

Reference solutions are those for which activity has been determined by comparison with one or more "absolute standard" solutions. This comparison may be made directly or through one or more intervening steps. Such solutions are far less expensive than the absolute standards, and if appropriate equipment and techniques are used, are almost as accurate ($\leq$ 1%).

Gaseous standards may be needed to calibrate counters used for monitoring gaseous radionuclides, such as tritium, $^{14}Co_2$ or ^{41}Ar, which are supplied in containers or small gas cylinders and are usually calibrated by direct counting.

In cases where the half-life of a desired radionuclide is short (e.g., ^{131}I with an 8.08 day half-life), "mock" standards may be used; i.e., long-lived nuclides with emission characteristics very similar to those of the short-lived nuclides. For example, a mixture of ^{133}Ba and ^{137}Cs, in a ratio of 12:1, suitably filtered to eliminate low-energy photons, simulates the photon spectrum of ^{131}I.

For quality assurance, and to assist the user, radionuclide standards are provided with a certificate of standardization, which includes the following data:

Principal radionuclide.
Form.
If liquid or gas, description (e.g., mass or volume and density at a given temperature chemical composition).
If solid, construction details and dimensions.
Activity and reference time.
Method of measurement.
Accuracy (uncertainty) of measurement.
Impurities.
Decay scheme.
Half-life.
Recommended handling procedures.

RADIOLOGICAL CALIBRATION AND STANDARDIZATION

Radiological standardization involves the measurement of radiation from a source, while determination of response of an instrument might be more properly termed "calibration."

(This same distinction holds true for radioactivity, although the two terms are often used interchangeably.) Radiological, as distinguished from radioactivity, refers to units of exposure, dose or flux rather than to the determination of activity or disintegration rate. Usually, radiological refers to standardization of sealed sources as opposed to "loose" or open sources.

Probably the oldest (and still a widely used) standard source of exposure is ^{226}Ra in equilibrium with its daughter products. In the ^{226}Ra decay scheme (Figure 11-8-3), equilibrium is reached within 30 days after the radium has been sealed into a capsule (ampoule).

Most of the gamma radiation associated with ^{226}Ra comes from two of its daughters, ^{214}Pb and ^{214}Bi. Fluorescence x-rays are also emitted, but these are of low energy and are mostly absorbed by the encapsulating material. The average (effective) photon energy of ^{226}Ra in equilibrium with daughters is 0.83 MeV. Standardization is best accomplished by weighing, and an uncertainty of < ± 1% can be obtained in this manner.

Radium sources are usually salts of radium ($RaCl_2$, $RaSO_4$) sealed in hollow needles or cylinders, usually of platinum about 0.5 mm thick. Some radium sources are sealed containers of solutions, usually of $RaSO_4$.

Radium-226 and other gamma sources can also be standardized or calibrated by direct measurement of the exposure rate by merely specifying exposure rate at a given distance. Disintegration rate of the source can be calculated from a complex equation involving all the different species of gamma photons emitted:

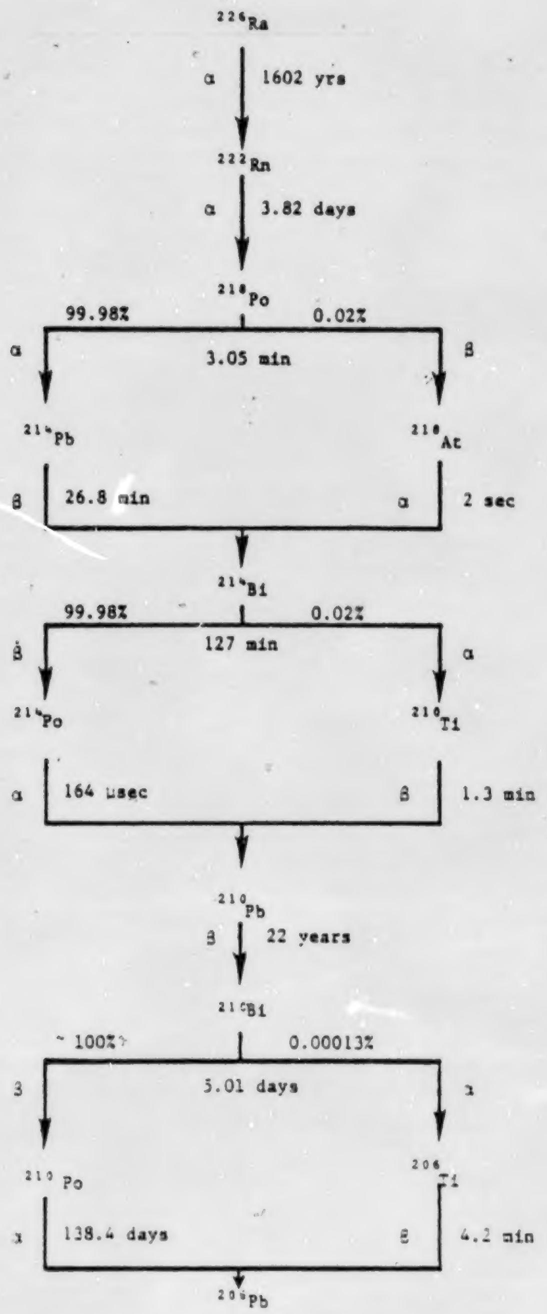
^{226}Ra
α
1602 yrs
^{222}Rn
α
3.82 days
^{218}Po
99.98%
0.02%
α
3.05 min
β
^{214}Pb
^{218}At
β
26.8 min
α
2 sec
^{214}Bi
99.98%
0.02%
β
127 min
α
^{214}Po
^{210}Tl
α
164 μsec
β
1.3 min
^{210}Pb
β
22 years
^{210}Bi
~ 100%
0.00013%
β
5.01 days
α
^{210}Po
^{206}Tl
α
138.4 days
β
4.2 min
^{206}Pb

Figure 11-8-3. Decay scheme for ^{226}Ra to stable ^{206}Pb, including half-lives and particle emission types.

$$R/hr = \frac{2.24 \times 10^{-9}\ S\Sigma_i n_i \mu_i E_i}{4\pi d^2 W} \qquad \text{Eq. 11-8-4}$$

where

Σ_i = summation of all terms.

S = source strength in photons per hour.

n_i = fractional yield of ith photon.

E_i = energy of ith photon, MeV.

μ_i = linear absorption coefficient for photons with energy E_i in air, cm^{-1}.

d = distance from source, cm.

W = average energy to produce one ion pair, taken to be 34 eV.

It is far more convenient to measure exposure with an ion chamber (See Module 11-2, "Gas-Filled Detectors I: Ionization Detectors.") and report source output in terms of specific gamma ray emission, Γ (Figure 11-8-4), which is in units of R/hr per mCi at a distance of 1 cm from source (assumed to be a point source). The measurement, of course, is made at a distance much greater than 1 cm to achieve good accuracy; however, accurate measurement of the exposure rate from any gamma source (or x-ray machine, for that matter) is difficult. Under the best of circumstances, an uncertainty of ± 2% is all that can be achieved.

If decay scheme is complex, or not well known, measurement of exposure is preferred method of calibration. A free-air ionization chamber should be used for rate measurements,

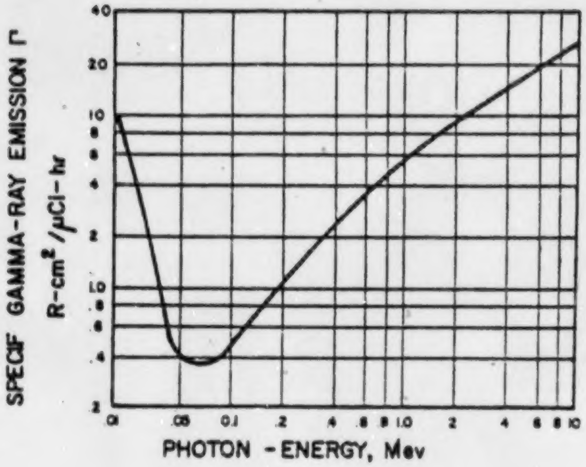

Figure 11-8-4. Specific gamma-ray emission, Γ, for a point source, assuming one photon of energy per disintegration.

and a small cavity chamber for measuring total exposure, which is fairly energy independent. In each case, "electronic equilibrium" must be achieved; that is, the number of electrons entering the sensitive volume must equal the number escaping from it. Because energies of secondary electrons are determined by gamma radiation energy, thicker walls are needed for higher-energy gamma rays; however, thick-walled chambers are not suitable for measuring low-energy photons because of absorption within the walls.

Alpha and beta sources are usually only standardized or calibrated in terms of activity; occasionally, however, beta source output is measured in terms of energy absorbed in air or rads. To accomplish this purpose, a very special kind of ionization chamber known as an "extrapolation chamber" is used. This chamber is adjustable in size so that a curve can be obtained of beta dose versus chamber depth, which can then be extrapolated to zero depth to provide the so-called surface dose.

Absolute calibration of neutron sources can be accomplished by calculation from activation foil data or by direct measurement. There are two commonly used methods: $MnSO_4$ bath and long counter. The $MnSO_4$ bath technique consists of immersing the neutron source in a large cylinder of $MnSO_4$ solution for a known period of time. A hollow, dry tube is provided in the $MnSO_4$ for the source. Neutron irradiation of the $MnSO_4$ primarily activates the ^{55}Mn, which is 100% of the naturally occurring Mn, to ^{56}Mn by the (n, γ) reaction. Then the Mn-56 with a half-life of 2.6 hours, is easily counted.

After a suitable irradiation period, usually about 16 hours to permit the activation process to reach equilibrium, the neutron source is removed and a NaI(Tl) crystal is inserted into the tube. The gamma rays from Mn-56 are then counted. Corrections must be made for absorption of gamma radiation in the bath, and for efficiency of the NaI(Tl) crystal. Activation of S and O does not occur; these elements both have low cross sections and neutron capture produces a stable nuclide.

The precision long counter (Figure 11-8-5) is a precisely made BF_3 tube in a specially constructed paraffin shield. The shield provides a response in the BF_3 that is independent of neutron energy. Neutron sources can be quickly and easily calibrated or standardized with the long-counter technique, with an uncertainty of only a few percent. Both $MnSO_4$ and the long counter provide a measure of neutron flux, which can be readily calculated:

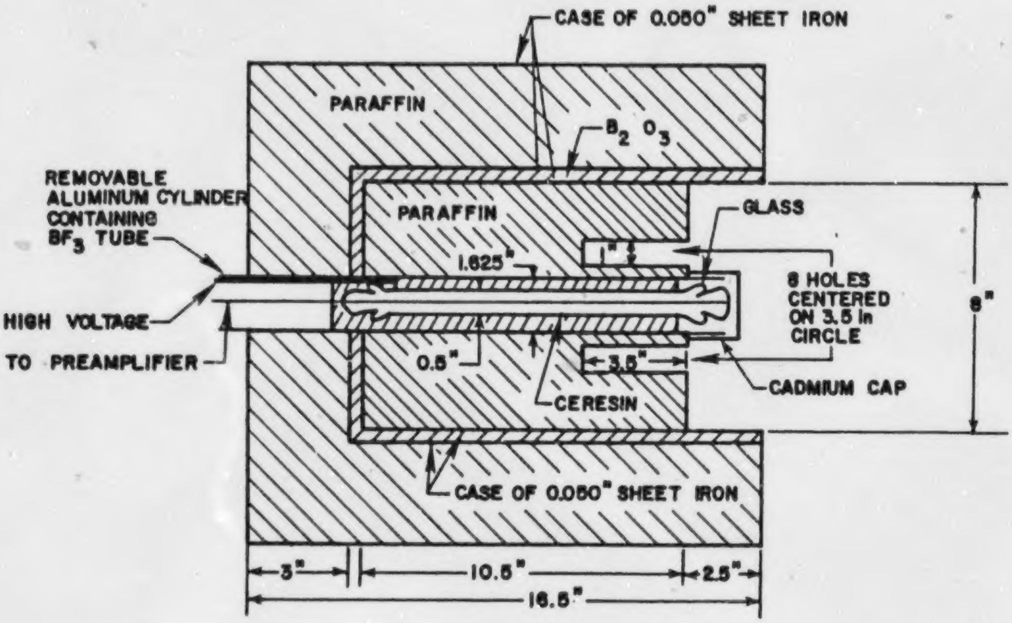
CASE OF 0.050" SHEET IRON
PARAFFIN
B_2O_3
REMOVABLE ALUMINUM CYLINDER CONTAINING BF_3 TUBE
PARAFFIN
GLASS
1.625"
1"
8 HOLES CENTERED ON 3.5 in CIRCLE
8"
HIGH VOLTAGE
TO PREAMPLIFIER
0.5"
3.5"
CADMIUM CAP
CERESIN
CASE OF 0.050" SHEET IRON
3"
10.5"
2.5"
16.5"

Figure 11-8-5. Precision long counter in cross section.

$$\frac{\Delta N}{\Delta t} = N\sigma\phi(1-e^{\lambda\tau})e^{-\lambda t} \qquad \text{Eq. 11-8-5}$$

where

$\frac{\Delta N}{\Delta t}$ = number of radioactive atoms decaying per second.

N = number of target atoms.

σ = cross section for ^{55}Mn (n,γ) ^{56}Mn reaction, cm^2.

ϕ = neutron flux, n/cm^2-sec.

λ = decay constant for radioactive species produced, seconds.

τ = irradiation time, seconds.

t = time between counting and end of irradiation.

The above equation can be rearranged to solve for ϕ:

$$\phi = \frac{\frac{\Delta N}{\Delta t}}{N\sigma(1-e^{-\lambda\tau})e^{-\lambda t}} . \qquad \text{Eq. 11-8-6}$$

Source strength, S, in neutrons per second, may be calculated:

$$\phi = \frac{S}{4\pi d^2} \qquad \text{Eq. 11-8-7}$$

where d = distance from source, cm.

Similarly, Equation 11-8-7 may be used to calculate source strength from flux obtained with the BF_3 counter.

Neutron flux is calculated from the long counter reading:

$$\phi = \frac{cpm}{kVN\sigma} \qquad \text{Eq. 11-8-8}$$

where

cpm = count rate, counts per minute.
k = constant equal to 60 seconds per minute.
V = sensitive volume of the BF_3 tube, cm^3.

N = number of atoms of ^{10}B per cm^3.

σ = cross section for $^{10}B(n,\alpha)\,^7Li$ reaction cm^2.

The preceding discussion has touched briefly on measurement of dose (as opposed to exposure, flux or activity). Calibration and standardization is seldom in units of dose, primarily because of the many practical difficulties associated with direct dose measurements. Calorimetry has been used for direct dose determination because, as ions recombine and excitations decay, absorbed energy gets converted to heat. However, any chemical changes induced in a calorimeter by radiation can cause errors in an absorbed dose reading. (Absorbed dose refers to whatever material is used in the calorimeter, which may not necessarily be tissue equivalent.)

Doses in liquids and in solid bodies can be measured by the "cavity ionization" technique. In this method, dose is deduced from the ionization in a small gas-filled cavity, which is essentially an ionization chamber with tissue equivalent walls. The cavity must be small enough not to affect secondary electrons produced by radiation in the liquid or solid. Energy deposition in the tiny cavity is identical to that in the liquid or solid surrounding it, and can be calculated:

$$E = JWS \qquad \text{Eq. 11-8-9}$$

where

E = energy, in MeV, deposited in cavity.

J = number of ion pairs.
W = energy, in eV, to produce an ion pair.
S = mass stopping power of medium relative to that of air.

The cavity chamber technique outlined above is also known as the Bragg-Gray principle: Bragg-Gray chambers are used for both beta and gamma dose measurements. Values of S (relative stopping power) are available in the literature, usually ranging from about 0.9 to 1.2, depending on energy and Z, as shown in Table 11-8-2.

PRACTICAL ASPECTS OF CALIBRATION

In the U.S., the National Bureau of Standards (NBS) serves as the primary standards agency and provides calibration of secondary sources of radiation. Gamma rays from ^{60}Co and ^{137}Cs are calibrated with cavity ionization chambers. A set of three free-air ionization chambers is used for standardization of exposure from x-rays in the 10 to 250 keV region.

Measurement uncertainties are ± 2% for x-ray exposures and less than ± 1% for gamma-ray sources. For very high-energy photons - those with energies above 3 MeV - a special calorimetric procedure is used that provides an uncertainty of about ± 1%. NBS also has two RaBe photoneutron sources with known neutron outputs and uncertainty of < 1%.

TABLE 11-8-2
MASS STOPPING POWER RATIOS, RELATIVE TO AIR, FOR MONOENERGETIC ELECTRONS

Electron kinetic energy	C	Water	"Tissue"
Mev			
0.001	1.079	1.254	1.231
.002	1.062	1.223	1.202
.003	1.055	1.211	1.190
.004	1.051	1.203	1.184
.005	1.049	1.198	1.179
.006	1.045	1.193	1.174
.007	1.045	1.192	1.173
.008	1.044	1.189	1.171
.009	1.043	1.188	1.169
.01	1.042	1.186	1.167
.02	1.037	1.177	1.158
.03	1.034	1.172	1.154
.04	1.033	1.169	1.152
.05	1.032	1.168	1.150
.06	1.031	1.166	1.148
.07	1.030	1.165	1.147
.08	1.030	1.164	1.146
.09	1.029	1.163	1.146
.1	1.029	1.162	1.144
.2	1.023	1.158	1.140
.3	1.019	1.155	1.138
.4	1.014	1.154	1.136
.5	1.010	1.152	1.134
.6	1.007	1.148	1.130
.7	1.003	1.144	1.126
.8	1.000	1.140	1.122
.9	0.997	1.134	1.116
1.0	.994	1.133	1.115
2	.971	1.105	1.088
3	.954	1.086	1.069
4	.942	1.071	1.054
5	.932	1.059	1.042
6	.923	1.049	1.032
8	.909	1.032	1.016
10	.898	1.019	1.003

NBS calibrates standardizing instruments and sources of various types. Secondary exposure-measuring standards, such as ion chambers (R-meter, transfer chambers), are usually calibrated with an uncertainty of ± 2 to 10%, depending on conditions such as exposure rate and selected energy range. Gamma-ray outputs from radium sources in the range from 0.5 to 250 mg, can be determined with an uncertainty of 0.5 to 1% by comparison with known sources.

Cobalt-60 sources can be calibrated with a somewhat greater uncertainty ($\leq$ 3%), and the range is limited to sources emitting 0.5 to 200 mR/hr at 1 meter. Neutron sources are calibrated (by comparison with a national standard source in a $MnSO_4$ bath) with an uncertainty in measurement of 2.5 to 5%, depending on source strength. NBS also provides calibrated alpha-, beta- and gamma-emitting nuclides in the microcurie range, calibrated by 4π or other absolute counting techniques, with uncertainties of 2% down to as low as ± 0.1%.

The striking feature of radiological and radioactivity measurements is their relatively large uncertainties, particularly when compared with other physical measurements such as mass, length, frequency or time. Table 11-8-3 shows the best available accuracies. Radiation measurements with an uncertainty of a few percent are considered quite good indeed; in the laboratory, 10 to 20% may be the rule.

Under field conditions, where a radiation field may not be well characterized (e.g., may be nonuniform, mixed, or contain scattered photons) and environmental conditions are uncontrolled, even higher uncertainties are common, as shown in Table 11-8-4. Compare these uncertainties with the accuracy of a watch, which may gain or lose a full min-

TABLE 11-8-3

ABSOLUTE METHODS FOR PREPARATION OF RADIOACTIVE NUCLIDE STANDARDS

Method	Nuclides to which method is applicable	Activity of measured source	Special corrections, limitations, etc.	Accuracy under most favorable conditions[*]
1. *4π Counting*				
(a) Geiger-Müller counter	α emitters, β emitters, with or without γ rays.	0.01μc or less	Accuracy usually limited by uncertainty in corrections for loss of β particles in source material and source mount, particularly for β particle energies <0.5 Mev.	±2%
(b) Proportional counter	do	do	do	±2%
(c) High pressure proportional counter.	Electron-capturing nuclides	do	Method assumes knowledge of K-fluorescence yield and K/L capture ratio.	±5%
(d) Scintillation counting; solid phosphor.	β emitters, with or without γ rays.	do	Suitable for medium and high energy β emitters; low energy cut-off limits accuracy for low energy β emitters.	±2%
(e) Scintillation counting; liquid phosphor.	β emitters, with or without γ rays, α emitters.	do	Suitable for medium and high energy β emitters; low energy cut-off limits accuracy for low energy β emitters.	±2%
2. *Defined Solid Angle*				
(a) End window Geiger-Müller counter.	β emitters	0.1μc or more	Complex corrections for absorption and scattering effects	±5%
(b) Proportional counter	α emitters	do		±0.5%
	β emitters	do		±5%
(c) Zinc sulphide screen	α emitters	do		±0.5%
3. *Coincidence Counting*				
(a) $\beta\gamma$	$\beta+\gamma$ emitters having simple decay schemes.	1μc or less	Uncertainty in the correction for the gamma sensitivity of the β detector usually limits accuracy.	±2%
(b) $4\pi\beta\gamma$	Many $\beta+\gamma$ emitters	0.1μc or less	Corrections, in general, very small and hence method is more accurate than $\beta\gamma$ method.	±1%
(c) $\gamma\gamma$	So far chiefly used for Co^{60}.		Requires careful adjustment of efficiencies of counters, and allowance for angular correlation of γ rays.	±1%
(d) $X\gamma$ and $4\pi X\gamma$	Electron-capturing nuclides	0.5μc or more		±5%
4. *Internal gas counting*	Low-energy β emitters Electron-capturing nuclides.	0.01μc or less	Limited to nuclide samples having suitable gaseous form	±2%
5. *Measurement of loss of charge.*	β emitters	0.1mc	Application limited by experimental difficulties	±5%
6. *Calorimetry*	α emitters	1mc or more		±0.5%
	β emitters	1mc or more	Involves a knowledge of the mean energy of the β particles.	±2%
	$\beta+\gamma$ emitters	1mc or more	Involves a knowledge of the mean energy of the β particles, the quantum energies of the γ rays, the number of γ quanta of each energy per disintegration and the fraction of the total γ radiation absorbed in the calorimeter.	±5%
7. *Ionization measurement of dose rate.*	γ emitters having relatively simple γ-ray spectra.	1mc or more	Accuracy of disintegration rate determination limited by uncertainty in W (energy per ion pair) for air and in some cases by incomplete knowledge of γ-ray spectra.	±1%
	β emitters		Accuracy limited by uncertainty in W and in average energy of β particles.	±5%
8. *Weighing*	Radium and long-lived radionuclides.	10 mg	High chemical and radio-chemical purity of nuclide or parent essential.	±0.2%[†]
9. *Mass spectrometer*	Has been used for H^3			±1% (routinely)

[*] The figures given in this column are estimates of the extreme limits of total error in the disintegration-rate value, and the errors which contribute to this total depend on various factors which are not the same for each nuclide; the limits given indicate the accuracy believed to be obtainable under the most favorable conditions presently available.

[†] Radium standards.

ute a day, and as such has an uncertainty each day of only ± 0.07%, or better than 1 part in 1400.

Under field conditions, as well as in the laboratory, many different sources and instruments are used. Instru-

TABLE 11-8-4
PRACTICAL ACCURACY OF STANDARD SOURCES AND MEASUREMENTS

Category	Radiation	Quantity	Practical Uncertainty*
Secondary	Alpha	Activity	± 2% (1 part in 50)
	Beta	Activity	± 2% (1 part in 50)
	X-Ray & Gamma	Exposure	± 2% (1 part in 50)
	Neutron	Neutron Yield	± 5% (1 part in 20)
Laboratory	Alpha	Activity	± 4% (1 part in 25)
	Beta	Dose	± 10% (1 part in 10)
	X-Ray & Gamma	Exposure	± 10% (1 part in 10)
	Neutron	Dose	± 20% (1 part in 5)
Operational (Field)	Alpha	Activity	± 15% (1 part in 7)
	Beta	Dose	± 25% (1 part in 4)
	X-Ray & Gamma	Exposure	± 20% (1 part in 5)
	Neutron	Dose	± 35% (1 part in 3)

* Uncertainty is taken to be $\pm 3\sigma$ and includes both systematic and random errors.

ments may be calibrated with any of a number of sources, but the source and calibration conditions should be nearly the same as those encountered under operating conditions.

An important calibration function relates to instruments used for radiation monitoring in connection with health physics or radiation protection. Table 11-8-5 lists alpha-emitting nuclides commonly used for this purpose, the most commonly used being ^{230}Th and ^{239}Pu, with ^{241}Am now enjoying greater use.

Many beta emitters are also used (Table 11-8-6). For example, ^{90}Sr-Y can be used for dose rate calibrations, though a slab of uranium metal is more commonly used. The beta dose rate at the surface of a slab of uranium is 233 mrad/hr and

TABLE 11-8-5

ALPHA-EMITTING NUCLIDES SUITABLE FOR INSTRUMENT CALIBRATION AND STANDARDIZATION

Nuclide	Half-life (yr)	Alpha Energy (MeV)	Percent Abundance
^{148}Gd	84	3.18	100
^{210}Po	0.38	5.305	100
^{230}Th	7.7×10^4	4.617	24
		4.684	76
^{234}U	2.35×10^5	4.717	28
		4.763	72
^{235}U	7.1×10^8	4.37	18
		4.40	57
		4.58	8
^{238}U	4.51×10^9	4.15	25
		4.20	75
^{238}Pu	87.8	5.456	28
		5.499	72
^{239}Pu	2.44×10^4	5.105	12
		5.143	15
		5.156	73
^{241}Am	433	5.442	13
		5.484	88
^{242}Cm	0.44	5.965	0.035
		6.066	26
		6.110	74
^{244}Cm	18	5.661	0.017
		5.764	23
		5.806	77

drops off quite rapidly so that, at a distance of only 4 inches, the dose rate is less than 10 mrad/hr.

Many gamma-emitting sources are available for instrument calibration and standardization. For spectroscopy, virtually any radionuclide can be obtained, but usually ^{137}Cs with its 0.662-MeV peak, and ^{60}Co, with peaks at 1.17 and 1.33 MeV, are used. These same two nuclides, along with ^{226}Ra + daughters, are major sources for instrument calibration. Low-energy exposure calibrations are usually achieved with x-ray machines.

TABLE 11-8-6

BETA-EMITTING NUCLIDES SUITABLE FOR INSTRUMENT CALIBRATION AND STANDARDIZATION

Nuclide	Half-life (yr)	E_{max} (MeV)	Percent	E_{avg} (MeV)	Significant Photons (MeV) and Yield
^{3}H	12.35	0.0185	100	0.0057	None
^{14}C	5730	0.156	100	0.045	None
^{45}Ca	0.45	0.257	100	0.077	None
^{85}Kr	10.76	0.674	99	0.22	0.514 (41%)
^{90}Sr-^{90}Y	27.7	0.546	100	0.18	None
(in equilibrium)		2.27	99+	0.93	
^{147}Pm	2.62	0.224	100	0.070	None
^{204}Tl	3.78	0.763	98	0.243	Hg K and L x-rays
^{210}Pb-^{210}Bi + daughters	20.4	0.016	85	~0.005	0.047 (4%)
		0.061	15	~0.020	
		1.161	100	~0.36	
^{Nat}U + daughters	4.51×10^{9}	0.100	35	~0.03	Many, including K and L x-rays
		0.191	65	~0.06	
		2.229	98	~0.72	

Also, a large number of neutron sources are available; most commonly, ^{238}PuBe or ^{239}PuBu are used for instrument calibration. A spontaneous fission source, ^{252}Cf, is becoming more common, although it is expensive and has a high gamma dose rate. Data on neutron sources commonly used are summarized in Table 11-8-7.

It should be noted that instrument and dosimeter calibration (and standardization) is an exacting task. To simplify that task, special calibration facilities, built to permit highly accurate and reproducible measurements or exposures,

TABLE 11-8-7
COMMON NEUTRON SOURCES SUITABLE FOR INSTRUMENT CALIBRATION AND STANDARDIZATION

Source	Reaction	Half-life	Average energy (MeV)	Approx. yield (n/sec-Ci)	Approx. neutron dose rate (mrem/ hr-Co at 1 m)
^{124}SbBe	(γ, n)	60 days	0.024	1.6×10^6	0.16
^{210}PoB	(α, n)	138 days	2.3	6.0×10^5	0.6
^{210}PoBe	(α, n)	138 days	4.5	2.5×10^6	2.9
^{226}RaBe	(α, n)	1640 yr	3.9	1.5×10^7	16.6
	(γ, n)	1640 yr	0.3	1.2×10^6	0.1
^{238}PuBe	(α, n)	87.8 yr	4.5	1.8×10^6	2.1
^{239}PuBe	(α, n)	24,400 yr	4.5	1.5×10^6	1.8
^{241}AmBe	(α, n)	458 yr	4.4	2.2×10^6	2.6
^{244}Cm	SF	18.4 yr	Fission	1.5×10^5	0.2
^{252}Cf	SF	2.65 yr	Fission	4.3×10^9	4.6×10^3

are often used. One such device (Figure 11-8-6) is designed for calibration or for exposing film or TLD dosimeters to known doses.

In summary, it might be well to consider the meaning of the word "calibration." It is used in many ways, but most often as follows: calibration is a process by which the value of a source is obtained, or by which the response or reading of an instrument, relative to some agreed upon standard value or series of values, is determined.

Calibration is a vital and important tool and, as such, is an integral part of any radiation detection and measurement operation.

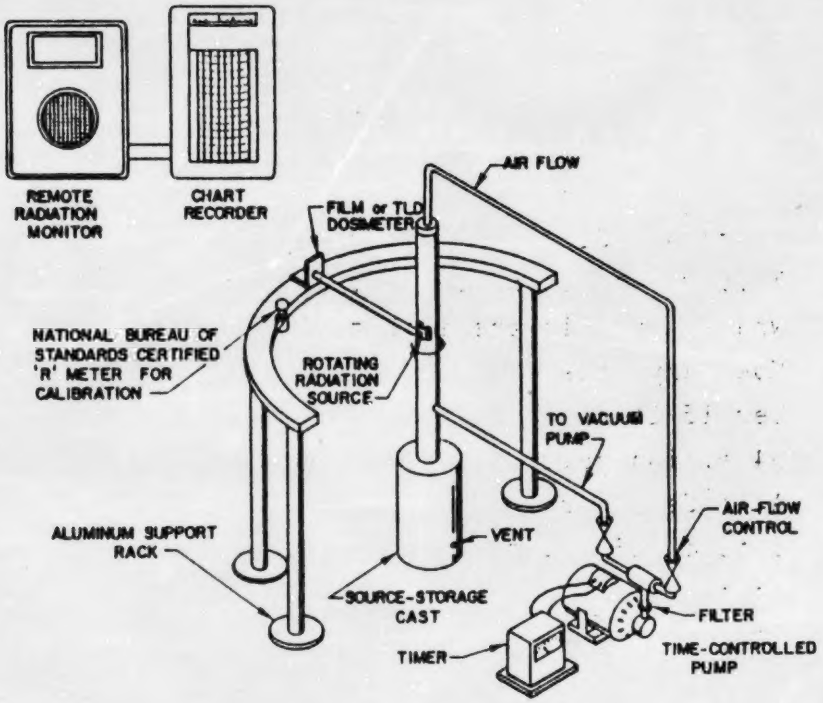
REMOTE RADIATION MONITOR
CHART RECORDER
AIR FLOW
FILM or TLD DOSIMETER
NATIONAL BUREAU OF STANDARDS CERTIFIED 'R' METER FOR CALIBRATION
ROTATING RADIATION SOURCE
TO VACUUM PUMP
AIR-FLOW CONTROL
ALUMINUM SUPPORT RACK
VENT
SOURCE-STORAGE CAST
FILTER
TIMER
TIME-CONTROLLED PUMP

Figure 11-8-6. Calibration setup for TLD or film dosimeter. Source is timer-controlled and rotates to ensure uniform irradiation. Source is raised by vacuum. Remote monitor and recorder provide a positive indication that source was properly positioned during entire irradiation period. (Adapted from IAEI)

EXERCISES

1. Convert Equation 11-8-3 so that it is solved for activity in units of Curies and Becquerels.
2. A reading taken on a TLD dosimeter is interpreted as 200 mrem with an uncertainty of ± 20% at 95% confidence. If this badge was worn by a physicist (who claims to have worked in a field of 100 mrem/hr for 2.5 hours), what chance is there that this claim is correct?
3. How many ions would be produced in a 1 cm^3 cavity chamber if it had received a dose of 10 rad? Assume $W = 34$ eV/ion pair and $S = 1.065$.
4. A ^{137}Cs laboratory standard of 10 μCi gave a net count rate of 1370 cpm. What is the counter efficiency? What kind of counter do you think this is? What is its accuracy and precision?
5. Read and summarize those parts of 10CFR that pertain to requirements for calibrating portable survey meters (all types) used in nuclear power plants.

MATERIALS

Alpha sources (known and unknown).
1 2π ionization counter.
1 gamma source.
Portable GM survey meters.
1 x-ray facility.

PROCEDURES

STANDARD SOURCE

1. Prepare an unknown alpha source for counting.
2. Count source in a 2π ionization counter.
3. Calculate activity of source.
4. What is efficiency of counter?
5. List possible sources of error.
6. What do you estimate the magnitude of error to be?
7. Specify accuracy and precision.
8. Count an alpha source of known activity in the 2π counter.
9. Calculate activity of source from counts; compare with known activity; explain any differences.

INSTRUMENT CALIBRATION - KNOWN SOURCE

1. Place standard gamma source at a known distance from GM survey meter; record the distance.
2. Record source type, activity and gamma energy(s).
3. Find specific gamma-ray emission, Γ.
4. Calculate dose rate corresponding to distance above.
5. Adjust survey meter until it reads same as value calculated above.
6. Repeat steps 1 through 5 above for another source of different energy; use a different survey meter.
7. Take a reading of second source with first survey meter. Explain any differences between the survey meters.

INSTRUMENT CALIBRATION - CONDENSER R-METER

Following Victoreen procedures, calibrate a GM portable survey meter for a particular x-ray facility in which a wide spectrum of x-ray energies are present.

DATA TABLE

To be devised by student.

REFERENCE MATERIALS

Campion, P.J. and Williams, A. "Standards of Radioactivity." *Health Physics* 11 (1965): 769.

International Atomic Energy Agency (IAEA). "Handbook on Calibration of Radiation Protection Monitoring Instruments." International Atomic Energy Agency Technical Reports Series, No. 133. Vienna: IAEA, 1971.

International Commission on Radiation Units and Measurements (ICRU). "Certification of Standardized Radioactive Sources." Report 12. Washington, DC: National Bureau of Standards, 1968.

________. "Radiation Quantities and Units," Report 19. Washington, DC: National Bureau of Standards, 1971.

Kathren, R.L. "Standard Sources for Health Physics Instrument Calibration." Health Physics 29 (1975): 143.

McNish, A.G. "Nomenclature for Standards of Radioactivity." International Journal of Applied Radiation and Radioisotopes 8 (1960): 145.

National Bureau of Standards. "A Manual of Radioactivity Procedures," Handbook 80. Washington, DC: National Bureau of Standards.

________. "Radiation Quantities and Units," Handbook 84. Washington, DC: National Bureau of Standards.

________. "Report of the International Commission on Radiological Units and Measurements (ICRU)," Handbook 78. Washington, DC: National Bureau of Standards.

Wang, Y., ed. Handbook of Radioactive Nuclides. Cleveland, OH: Chemicals Rubber Co., 1969.

Wildback, W.A.; Powell, R.C.; and Mason, H.L. "Accuracy of Measurements and Calibrations," Technical Note No. 262. Washington, DC: National Bureau of Standards.

TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. What is radioactivity standardization? How does it differ from radiological standardization?

2. What is an absolute standard?

3. Answer these questions about 4π counting:

   a. What does 4π counting measure?

   b. For what types of emitters is 4π counting useful?

   c. What is the efficiency of the 4π counter?

4. Briefly describe mass determination as a method of standardization of a radioactive source.

5. What is calorimetry? ______________________________

____

____

6. Answer these questions about ionization measurement:

   a. What is the purpose of ionization measurement?

   ____

   ____

   b. In order to use this method, what two values would you need to know?

   ____

7. What is the difference between an absolute standardized solution and a reference standard solution?

____

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8. Radionuclides are provided to the user with a certificate of standardization. List four of the items (types) of information about the radionuclide given in the certification report.

   a. ____

   b. ____

   c. ____

   d. ____

9. What is the purpose of the long counter and MnS04 bath?

10. Briefly describe the cavity ionization (Bragg-Gray) technique, including its purpose and common uses.

11. Explain how a calibrated instrument can be used to calibrate a source.

12. Under what circumstances would measurement of exposure be the preferred means of calibrating a gamma source?

13. What is a mock standard? When would a mock standard generally be used?

14. What is the difference between the accuracy and the precision of an instrument?

15. Briefly explain what is meant by traceability?